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HISTORY

# History of the Engineering Technology Division Oak Ridge National Laboratory 1944 – 1992

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Compiled by

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Prepared by
OAK RIDGE NATIONAL LABORATORY
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## DEDICATION

This history is dedicated to current and former staff members of the Engineering Technology Division and all its antecedent organizations. It is our intent that this document recognize and honor your contribution to the development and implementation of technology for the betterment of our society and all mankind.

J. E. Jones Jr.

## CONTENTS

Page

## ACKNOWLEDGMENTS

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# 1. INTRODUCTION

The early beginnings of the Engineering Technology Division (ETD) are inextricably enmeshed with the 1943 establishment at Oak Ridge of a pilot plant complex for the Hanford Engineering Works at Hanford, Washington, where plutonium was to be produced. This pilot plant complex had a fourfold objective: (1) to produce small amounts of plutonium, (2) to develop a chemical process for separating and purifying the product, (3) to evaluate the health hazards associated with processing and handling large amounts of highly radioactive materials from a uranium reactor, and (4) to train personnel to operate the full-scale plutonium production plant.

The pilot plant complex (the Clinton Laboratories), to be operated by the Metallurgical Laboratory, Manhattan District, of the University of Chicago, was built by the E. I. duPont de Nemours Company, Inc. Construction was begun on February 1, 1943; the uranium-metal-fueled, graphite-moderated reactor for plutonium production (the X-10 Graphite Reactor) was placed in operation on November 4, 1943. The second major facility constructed was the chemical pilot plant where the process for separating and purifying plutonium was to be developed. Other buildings constructed at this time were laboratories for chemistry, physics, and medical (health physics) research; machine shops; instrument shops; and several administrative buildings and warehouses.

The Clinton Laboratories was staffed by personnel from the Technical Division of the Metallurgical Laboratory at the University of Chicago, the duPont Company, and the Army. The duPont personnel were to provide plant operation experience and to transfer experience gained at Clinton Laboratories to the Hanford Engineering Works operated by duPont.

The organization included a medical division with health physics and biology groups, a chemistry division, a separations development division, an analytical division, an engineering development section, and a physics division. In succeeding years the Engineering Development Section evolved into the Technical Division, the antecedent of ETD.

The Engineering Development Section was headed by M. C. Leverett. Early members of the section included S. E. Beall, R. B. Briggs, M. D. Peterson, W. A. Rodger, A. F. Rupp, M. D. Silverman, and J. T. Weills. D. G. Reid and E. J. Witkowski, from the duPont Company, and O. Sisman and B. Manowitz, from the Army's Special Engineering Detachment, were also members of the section. The first two became Clinton Laboratories employees in March 1944, while the latter two made the transition in March 1946. A. F. Rupp and S. E. Beall were among those who moved to Hanford as duPont employees but returned to Clinton Laboratories later.

In 1944, J. R. Huffman also joined the section. R. N. Lyon, J. A. Kyger, J. E. Cunningham, and R. Van Winkle joined in 1945. B. Manowitz resigned in August 1946 to enroll at Columbia University.

From these, M. C. Leverett, M. D. Peterson, R. B. Briggs, and S. E. Beall would each become division director during the ensuing years. R. N. Lyon would become associate division director, and J. R. Huffman, J. A. Kyger, D. G. Reid, and W. A. Rodger became section heads. A. F. Rupp became ORNL Laboratory Services Superintendent.

The section was primarily concerned with Hanford problems and improvement of Clinton Laboratories operations. In addition to designing and testing equipment, the technical personnel of this section also planned to contribute to the design of future reactors. In its role of providing technical engineering assistance, the two primary activities pursued were (1) collaborating with physicists and metallurgists to upgrade the performance of the graphite reactor and (2) providing information for use in building the water-cooled reactors at Hanford. (The performance upgrade efforts resulted in an increase in power from 1000 to 4000 kW.) The section also collaborated with development and research groups on special projects.

By the fall of 1944, the primary assignments of the Engineering Development Section either had been completed or were near completion. Also, pilot plant personnel who were trained to staff the Hanford processing plant were being transferred to that facility. In October, the remaining staff of the Separations Development Division that supported operation of the chemical processing plant and the Engineering Development Section were combined to form the Technical Division of Clinton Laboratories. The new division had about 50 chemists and engineers; M. C. Leverett was division director.

F. L. Steahly, who would become the first director of the Chemical Technology Division, became a member of the Technical Division as a consequence of the merger, as did J. A. Lane (who would also become a division director) and F. C. McCullough. The three joined Clinton Laboratories in 1943.

J. A. Lane resigned from Clinton Laboratories in the fall of 1944 to go to Germany to assist the military and others in determining the state of nuclear energy and weapons development achieved by Nazi Germany. He again became an employee of what was then named the Clinton National Laboratory in January 1948.

Through successful achievement of the wartime objectives, the importance of a broad program on nuclear energy for both peaceful and military applications was firmly established. Transition from plutonium production, process development, and testing was begun in December 1944. The reactor and the chemical pilot plants were converted from plutonium production operations to experimental use at that time.

During the first half of 1945, scientists at Clinton Laboratories turned their attention to identifying important research and development (R&D) activities to be pursued for making the greatest contributions to the new field of nuclear science and technology. Fundamental research activities were clearly seen as needed. Also important were needs arising from the requirements of weapons development groups and from the desirability of obtaining basic information for reactor design.

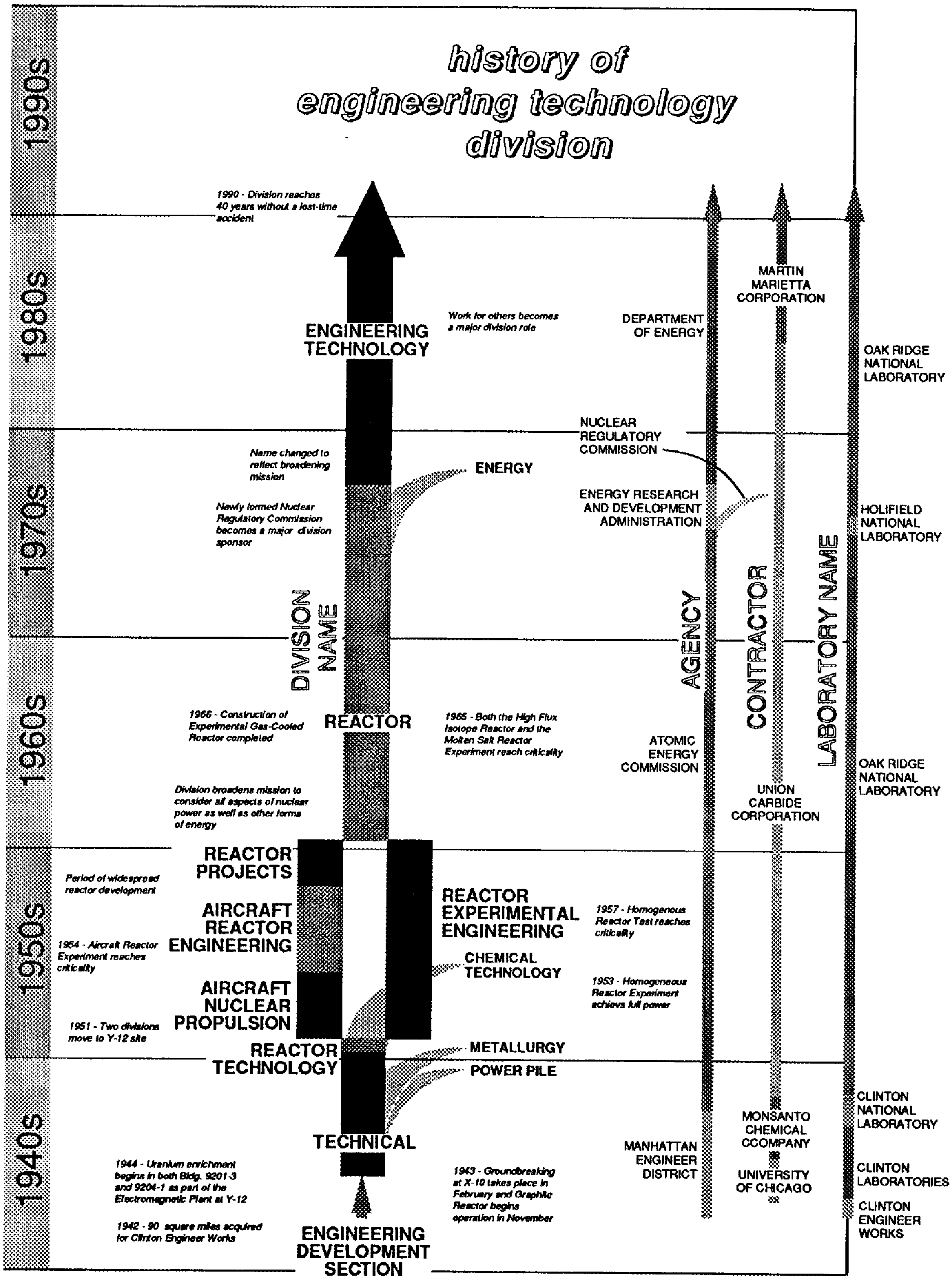
All initial objectives for which the Clinton Laboratories had been established were successfully accomplished by June 1945. With its responsibilities at the Clinton Laboratories under the Manhattan District fulfilled, the University of Chicago withdrew as operator. The Monsanto Chemical Company assumed the operating responsibility, effective July 1, 1945. Under Monsanto, R&D work on the design of a high-flux experimental reactor and larger scale preparation of radioisotopes for special use within the Manhattan Project were pursued together with basic research in physics, chemistry, and biology.

The purpose of this report is to describe the evolution and contributions of ETD from its beginning in 1944 up to 1992. To aid in understanding the evolution from Technical Division to ETD, changes are graphically depicted on p. 3. Key personnel, including division directors; associate, assistant, technical and deputy directors; section or department heads; and program or project leaders are listed in Tables A.1 and A.2 of Appendix A. Division directors are shown on p. 4.

Note that in 1951 the division (Reactor Technology Division at that time) was divided into two—the Reactor Experimental Engineering Division and the Aircraft Nuclear Propulsion Division. The two were recombined to form the Reactor Division in 1960. In the interim, the Aircraft Nuclear Propulsion Division underwent two name changes.

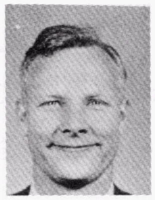
The timeline and Tables A.1 and A.2 give an overview of the division evolution and the leaders responsible. With this background, the reader can better appreciate the scope of work and achievement by the organization now titled "Engineering Technology Division."

ORNL-DWG 92M-3386 ETD



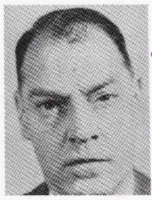
## DIVISION DIRECTORS

M.C. Leverett
1944-48



M.D. Peterson
1948-49

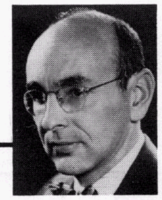
A.M. Weinberg
1949-51



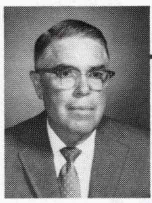
R.C. Briant
1951-54

S.J. Cromer
1954-58

C.E. Winters
1951-53

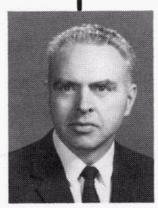


J.A. Lane
1953-58



W.H. Jordan
1958-59

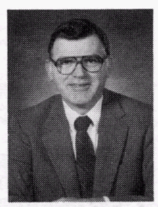
R.A. Charpie*
1959-61



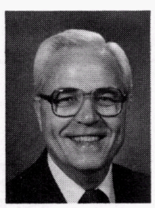
R.B. Briggs
1958-60

H.G. MacPherson
1961-63

S.E. Beall
1963-74



G.G. Fee
1974-78

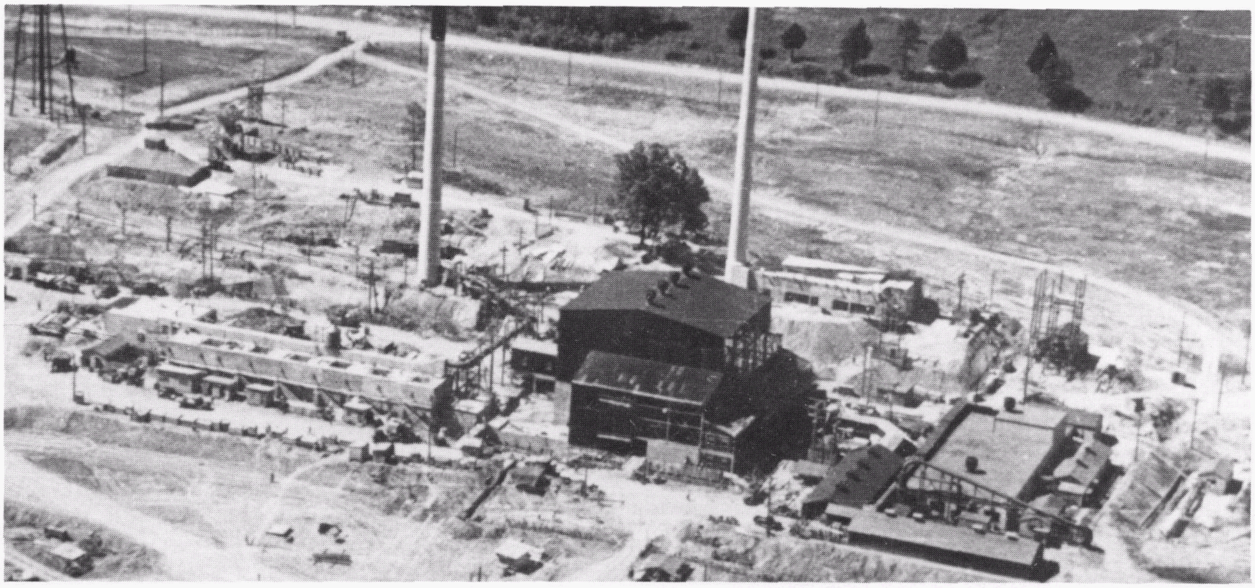


H.W. Trammell
1978-89

J.E. Jones Jr.
1989-

*Also first director when divisions were recombined.

*The Oak Ridge Graphite Reactor was built in 1943 as a part of the pilot plant complex (the Clinton Laboratories) for the Hanford Engineering Works at Hanford, Washington, where plutonium for atomic weapons was to be produced. The pilot plant complex was to provide small amounts of plutonium to allow development of a chemical process for separating and purifying the product. The reactor was placed in operation on November 4, 1943.*



*Early view showing the pilot plant complex at Clinton Laboratories for producing plutonium. The dark building (Building 105, now 3042) near the center houses the Graphite Reactor (X-10 Pile). The building to the left of 105 is Building 205, the separations building.*

*Front-face view of the Graphite Reactor where workmen are removing fuel slugs by pushing them into a channel at the back of the reactor.*

*Building 205 (now Building 3019) housed the companion chemical pilot plant where the process for separating and purifying plutonium was to be developed to complete the overall production and separation complex of Clinton Laboratories. All equipment for the operations was enclosed in "hot cells" surrounded by 5-ft-thick concrete. Remote control was required for even the simplest operations.*

## 2. 1944–1951 THE EARLY YEARS (Technology/Reactor Technology Division)

Because of the unique facilities for pilot plant and research work as well as the unusual talents of the staff, the successful accomplishment of the original four objectives for which Clinton Laboratories was established did not mark the end of operations as initially expected. Rather, the Clinton Laboratories embarked on a new course that included addressing fundamental research and engineering development associated primarily with nuclear reactor design.

Consequently, in the year following the end of World War II, the Clinton Laboratories expanded its program and activities. The research program went through a transition from applied war research to a more balanced program of fundamental and applied research coordinated with the overall atomic energy program.

A major effort was devoted to research and development (R&D) work leading to the design of a high-flux experimental reactor and to larger scale preparation of radioisotopes for special uses within the Manhattan Project, as well as to basic research in physics, chemistry, and biology. The Technical Division was engaged in the first two and in the development of chemical processes.

Reactor development activities had been initiated late in 1944 by the chemists. They proposed the construction of a homogeneous reactor using an aqueous fuel solution containing enriched uranium and plutonium. Such a reactor was to be a research tool for preparation of large quantities of radioactive tracers and radiation sources. These would be used for studies of chemical radiation effects at high power levels and for the accumulation of data on operating characteristics, chemical stability, and general feasibility of homogeneous reactors. It was also thought that the aqueous fuel solution could be utilized very effectively for chemical processing studies and that the high neutron flux of the reactor would be useful for irradiating thorium in connection with studies of the preparation and extraction of uranium-233.* Uranium-233 was of considerable interest because of indications that the ratio of neutrons emitted to neutrons absorbed was higher than that for either uranium-235 or plutonium-239.

Physicists were also interested in the homogeneous reactor as a facility that would provide a high neutron flux for various experimental uses. Of particular interest was the desirability of studying or demonstrating the process of breeding†,‡ and possibly establishing a breeding cycle that would create more uranium-233 than was consumed in the reactor.

Work on homogeneous reactor design was pursued through 1945. However, at the end of that year, several major problems had not been solved. Possibly the most serious of these was the formation of bubbles in the homogeneous solution. Projections showed that, under certain conditions, it might be possible to set up undamped power oscillations, which would increase in magnitude until the reactor was out of control. Operating at elevated temperature and pressure to minimize the

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*Only uranium-233, uranium-235, and plutonium-239, which are fissile species, have sufficient stability to permit long-time storage. They are also fissionable by neutrons of all energies. Of these, uranium-235 is the only one that occurs in nature. The other two are produced artificially by bombardment of fertile species (thorium-232 or uranium-238) with neutrons; thorium-232 is used for production of uranium-233, and uranium-238 is used for plutonium-239 production.

†Breeding is generally considered to mean producing fissile species from fertile species. More specifically, breeding occurs when a fertile species (thorium-232 or uranium-238) is used to produce a fissile species that is the same as that used for the fuel in a reactor. When a reactor uses uranium-235 for fuel and uranium-238 as the fertile material to produce plutonium-239, it is called a converter.

‡Breeding and the development of a breeder reactor were major goals early in nuclear energy development because of the perceived scarcity of the uranium-235 isotope.

bubble problem, although potentially effective in overcoming this factor, was not considered seriously. Acceptable tank materials were not strong enough to sustain elevated temperatures and high pressures. Due to lack of experience in handling radioactive materials under pressure, constructing a completely new type of reactor to operate under high pressure was not an attractive alternative.

Additional major problems, associated with corrosion, solution stability, and large external holdup of fissionable material, remained unsolved at the end of 1945. Because resolving these problems was expected to entail extensive R&D work without assurance of success, the decision was made to pursue a heterogeneous reactor proposed earlier by physicists at the University of Chicago Metallurgical Laboratory.

The engineering design and development of this high-flux experimental reactor became a major effort at the Clinton Laboratories. A heterogeneous reactor design evolved that made use of light water as the moderator and coolant (following advice from E. P. Wigner*): a beryllium reflector and fuel elements in the form of flat, aluminum-clad, uranium-aluminum (U-Al) alloy plates. The reactor, which was in the preliminary design stage in late 1946, appeared to be the final choice.

Thus, there existed in 1946 the essential design of the Materials Testing Reactor (MTR) to be built at the National Reactor Testing Station in Idaho. The design was similar to the original Metallurgical Laboratory uranium-233 converter of 1944. This reactor was destined to become a singularly important facility for studying irradiation effects on materials and for radioisotope production.

*E. P. Wigner, mathematical physicist, was at the Metallurgical Laboratory, Manhattan District, University of Chicago, from 1942 to 1945; Director of Research and Development at Clinton Laboratories from 1946 to 1947; and Director of the Civil Defense Research Project, ORNL, from 1964 to 1965.

The Manhattan District assigned the work on the reactor to Kellex Corporation as design contractor. Monsanto Chemical Company, the contractor for operating the Clinton Laboratories, was the construction manager. In the fall of 1947, the design was considered to be sufficiently advanced to allow the first actual construction drawings to be finished by the end of the year.

As the work on the high-flux experimental reactor was being vigorously pursued, the radioisotope production program was expanding simultaneously. By July 1946, production capacity was sufficient to justify making radioisotopes available to users outside the Manhattan District installations. On August 2, 1946, the first radioisotope shipment was made under the radioisotope distribution program. The radioisotope development program continued in the Technical Division until it was transferred to the Operations Division in 1947.

The U.S. Army decided to support a Clinton Laboratories project based on a proposal by Farrington Daniels† of a conceptual design for a reactor‡ that could be operated at high temperature to produce power. The Technical Division initiated preliminary work on identifying problems and establishing a possible schedule for the Daniels Pile in January 1946. Those involved quickly determined that a large effort was required; therefore, the Power Pile Division was established, with C. R. McCullough as Director, to carry out the needed work.

Work on separation processes was divided between the Technical and Chemistry Divisions. The theory of solvent extraction was extensively studied and expanded, and solvent extraction columns were designed, constructed, and tested.

†Farrington Daniels, physical chemist, was on the faculty of the University of Wisconsin, Madison, from 1920 to 1959; on the staff of the Metallurgical Laboratory, Manhattan District, University of Chicago from 1944 to 1946; and he was a consultant to Clinton Laboratories in 1946.

‡This was a gas-cooled reactor. Although the reactor plant was never built, many of its design features were incorporated in later gas-cooled power reactor designs.

Because of the large quantities of enriched fissionable material to be processed, the separation process was recognized as being of fundamental importance to the operation of proposed experimental and power piles. Both the removal of fission products from the fuel and the separation and purification of other materials produced were to be included in the separation process development. Work on the design and development of these processes was pursued along with reactor development. Analytical methods also were developed for the detection of heavy elements encountered in chemical research and process development.

An Engineering Research Section was established with R. N. Lyon as Section Head. Work was begun on liquid metal heat transfer and led to significant advances in heat transfer and fluid flow analyses. R. N. Lyon was editor of *Liquid Metals Handbook* (2nd ed.); he also developed a heat transfer correlation that has seen perennial use.

Applied metallurgical research was introduced to address materials problems in reactor technology. Metallurgical work was done in connection with the preparation of U-Al fuel assemblies of the proposed high-flux experimental reactor. Some work was also done on the use of thorium and its alloys in control rods, and aluminum alloys were tested for water corrosion resistance under simulated reactor operating conditions. A separate Metallurgy Division was formed in 1946*.

Atomic energy activities were transferred from the Manhattan District to the newly formed Atomic Energy Commission (AEC) on January 1, 1947. The name of Clinton Laboratories was changed to Clinton National Laboratory to reflect the new permanent status of that organization under the AEC. Consideration was being given, at that time, to the long-range R&D activities and the size and composition of the scientific staff of the Laboratory. The AEC objective was to ensure that R&D activities placed at the national laboratories would avoid duplication of effort, and each laboratory would pursue those lines of effort for which it was best qualified.

In December 1947, the AEC announced plans to consolidate reactor development activities at the Argonne National Laboratory (ANL) near Chicago and to maintain the Clinton National Laboratory as a strong center for basic research, applied chemical research, and isotope production. As a part of this overall plan, efforts on the high-flux experimental reactor and on the power reactor were to be transferred to Argonne along with most of the technical people who had been carrying out these programs at Clinton National Laboratory. In anticipation of this fundamental change, the Kellex Corporation, in November 1947, was requested by the AEC to cease all work on the high-flux experimental reactor.

In keeping with the plans announced, the Power Pile Division was transferred to Argonne. This division would eventually join with Westinghouse Electric Corporation to develop the pressurized-water reactor (PWR) that is used in nuclear power plants today.

The decision to move the reactor work to Argonne resulted in changes in the Technical Division. M. C. Leverett left the Laboratory to become Research Associate at Humble Oil and Refining Company, and M. D. Peterson became division director. An organization chart for the Technical Division just before the 1948 departure of M. C. Leverett is provided in Appendix B. Some of the reactor design staff, including J. R. Huffman, W. A. Rodgers, and J. T. Weills, transferred to Argonne in 1947 and 1948. J. R. Huffman later would become an employee of the Phillips Petroleum Company, which was selected to be

*Seeing an important need, E. P. Wigner directed that a metallurgy division be established to do basic and applied work. The Metallurgy Division was therefore established in September 1946 in name only. In 1948, personnel in the Engineering Materials Section of the Technical Division were transferred into the Metallurgy Division. Although the Engineering Materials Section was shown on the organization chart of the Technical Division in 1949, it was only a shell organization.

operating contractor for the high-flux experimental reactor.

However, moving the high-flux experimental reactor development work proved impractical. As a result, Clinton was made responsible for design and development of all the reactor and other facilities inside the outer surface of the biological shield, and Argonne was given responsibility for the rest of the plant. The reactor became known as the MTR.

There was a change in Laboratory operators on March 1, 1948; Carbide and Carbon Chemicals Company (predecessor to Union Carbide's Nuclear Division) assumed the operating responsibilities. In addition, the name of the Laboratory was changed to Oak Ridge National Laboratory (ORNL).

ORNL's reactor development efforts were concentrated entirely upon the design of the MTR to be constructed at the new National Reactor Testing Station in Idaho. Design and development was continued until the 30-MW(t)* reactor was completed in 1952. The basic design developed at ORNL for the high-performance MTR has proved to be very flexible and to possess numerous advantages. A continuing program at ORNL was pursued not only to improve this reactor type, both in utility and operation, but also to facilitate the design of similar reactors.

As a part of the MTR development and design program, a full-scale mock-up of the reactor tank and major core components was constructed at ORNL for the performance of hydraulic tests to ensure that the design provided adequate cooling for the reactor core. When hydraulic experiments were completed and had demonstrated the adequacy of the design, ORNL had a full-scale mock-up of the MTR, with a cooling system and all of the basic features necessary for an operating reactor.

Authorization was obtained from the AEC to perform critical experiments in the MTR mock-up to check out the nuclear characteristics of the reactor design. A small amount of beryllium was used to simulate the beryllium reflector of the MTR, and fuel assemblies and a simplified control system were used for the critical experiments. These experiments demonstrated that the reactor would perform as planned.

The mock-up at this time had all the essential ingredients of a reactor. AEC authorization was requested and granted to provide shielding around the mock-up and to make other modifications necessary to operate it routinely as a research reactor. Provisions were made in the shield for experiments, and operation at power levels to 1000 kW was authorized by the AEC. [The nominal reactor power was later set at 3 MW(t).]

This Low-Intensity Test Reactor (LITR) first became critical in February 1950 and was subsequently placed in operation. It offered the highest neutron flux available and, therefore, was a particularly valuable addition to the ORNL research and radioisotope production facilities. The beautiful blue glow of Cerenkov radiation surrounding the fuel of an operating water-cooled and -moderated reactor was seen and photographed for the first time in this reactor.

M. D. Peterson left ORNL late in the summer of 1949 to become professor and Head of the Department of Chemistry at Vanderbilt University. A. M. Weinberg became Director of the Technical Division in addition to being Associate Director of ORNL. Earlier in 1949, A. M. Weinberg had suggested that a new look be taken at aqueous homogeneous reactors in light of work that had been done since 1945. The initial results looked very promising for further development, and by July the decision was made to establish a small development program.

*Reactors are generally rated by output in megawatts (MW), with a megawatt being a million watts. Thermal output from a reactor system is designated by MW(t); in the case of electrical generation, the electrical output is expressed as MW(e).

The AEC, by the end of 1949, had given approval for an R&D program leading to the construction of a homogeneous reactor. A major compelling feature of this type of reactor is that it makes it possible to incorporate into the system a chemical processing plant for treating the nuclear fuel on a continuous basis as opposed to being treated through batch operations. In mid-1950, ORNL was authorized to construct a pilot model reactor of the homogeneous type.

Another project that was assigned to ORNL was work on a reactor for nuclear propulsion of aircraft. In 1946, the U.S. Air Force (USAF) awarded a contract to Fairchild Engine and Airplane Corporation that established Fairchild as the responsible directing agency of a Nuclear Energy for Propulsion of Aircraft (NEPA) Project. The purpose of this project was (1) to perform feasibility investigations and research leading toward the adaptation of nuclear energy to the propulsion of aircraft and (2) to educate the aircraft engine industry concerning the field of nuclear science and its adaptation to aeronautical propulsion. M. C. Leverett was Technical Director of the NEPA Project from 1949 to 1951.

From mid-1946 until early 1948, the NEPA Project in Oak Ridge, Tennessee, was the only activity devoted to investigating and developing nuclear-powered aircraft technology. In 1948, the AEC asked the Massachusetts Institute of Technology (MIT) to investigate the feasibility of nuclear-powered flight. MIT sent scientists to Lexington, Massachusetts, for appraisal (i.e., the Lexington Project).The members of the Lexington Project concluded, in part, that there was a strong possibility that some version of nuclear-powered flight could be achieved if adequate resources and competent manpower were put into the development and that a vigorous and realistic aircraft reactor development program during the next few years should contribute to and benefit from other aspects of the Reactor Development Program of the AEC.

The Lexington Project report also recommended that a strong development program on nuclear-powered flight be undertaken if the decision was made that, as a national policy, the high cost in technical manpower, fissionable material, and money could be justified. From a military point of view, the cold war between the United States and Russia was a driving factor for exploring nuclear aircraft propulsion for long-range applications. (The nuclear aircraft propulsion work was eventually canceled in favor of support for intercontinental-ballistic-missile development.)

On April 27, 1949, a high-level conference was held at Oak Ridge to consider the part that ORNL could play in the Aircraft Nuclear Propulsion (ANP) Program. As a result of this meeting, the decision was made that ORNL would submit recommendations to AEC regarding participation.

Following numerous conferences between ORNL and AEC representatives, ORNL, in September 1949, gave written notice to AEC concerning its willingness to accept proposed ANP responsibilities and to carry out the program to the best of its ability with a priority second only to that of the MTR Project.

Having accepted the responsibility for the nuclear aspects of the ANP Program, ORNL established an ANP project and made plans for assembling suitable groups of R&D personnel. A major initial activity was the establishment of a Technical Advisory Board (TAB) of outstanding scientists who would meet at ORNL during the summer of 1950 to evaluate the various aircraft reactor designs under consideration and attempt to establish certain basic design points from which an aircraft reactor could be developed.

At the conclusion of the TAB meeting, a report was issued reflecting optimism for achieving supersonic flight. The Board recommended that an experimental aircraft reactor be constructed in Oak Ridge as soon as possible. Further, it recommended that a primary emphasis be given to a high-temperature liquid-cooled reactor. The AEC and the USAF decided at once to follow the TAB recommendations.

With the initiation of a major ANP effort at ORNL, the USAF program at the NEPA Project was closed out. Most of the technical groups were transferred to ORNL. However, several NEPA staff members joined a new division established by the USAF at the General Electric Company (GE) jet engine plant at Cincinnati to work on the much larger and longer range job of building a full-scale aircraft power plant fully integrated with jet engines. Instead of working, as expected, on the high-temperature liquid-cooled concept as its primary effort, this organization would later advocate an air-cooled reactor concept.

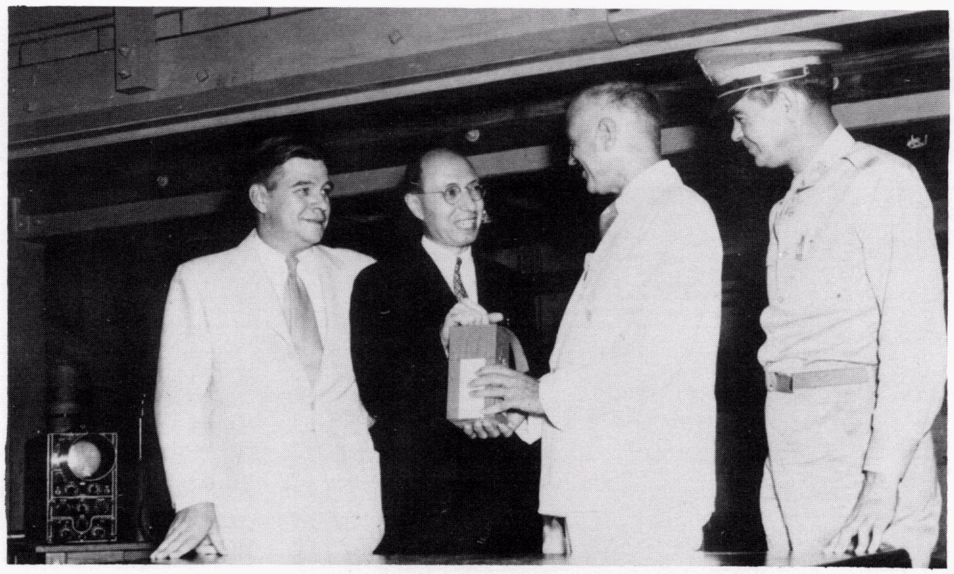
This change was supported by former NEPA employees, including M. C. Leverett, who were proponents of the air-cooled reactor concept. M. C. Leverett was Manager of Engineering of the Aircraft Nuclear Propulsion Department of GE from 1951 to 1956 and of the Development Laboratories of GE from 1956 to 1961.

The ORNL practice of building a reactor experiment was considered especially well suited for the aircraft reactor project. With the optimistic TAB report, ORNL obtained AEC support and initiated design and construction of an aircraft reactor experiment (ARE). (Simultaneously with most of the ORNL involvement, both GE and Pratt and Whitney were engine manufacturers participating in the overall Nuclear Propulsion Program.)

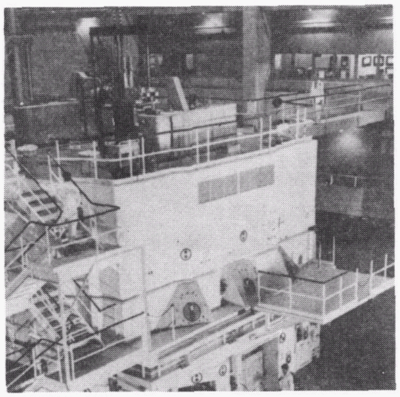
The prime effort was to address the basic feasibility problems associated with removing heat from a high-power density reactor operating at temperatures $\gtrsim 1500°F$. Heat would be removed by a liquid coolant with excellent heat transfer and transport properties and conveyed to air via heat exchangers located between the compressors and turbines of a set of jet engines. The design, construction, and testing of an ARE would be carried out in the ensuing 4 years.

By mid-1950, the Technical Division programs included a large effort in support of the MTR, an emerging effort on homogeneous reactors, and a large effort on chemical processing. In addition, ORNL was to undertake the new ANP Program, with heavy emphasis on work in the Technical Division. On July 1, the Technical Division was divided into a Reactor Technology Division and a Chemical Technology Division. A. M. Weinberg was director of the first, which embraced the three reactor programs; F. L. Steahly was named director of the second.


***Newspaper headlines on August 7, 1945, the day after the atomic bomb was dropped on Hiroshima.***

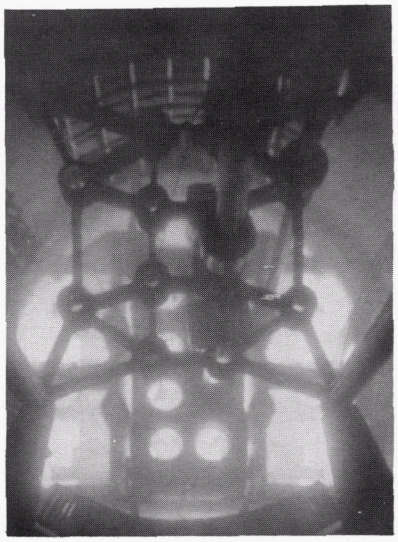


***First isotope shipment in August 1946. E. P. Wigner hands carbon-14 (used in biological research) container to recipient from St. Louis.***

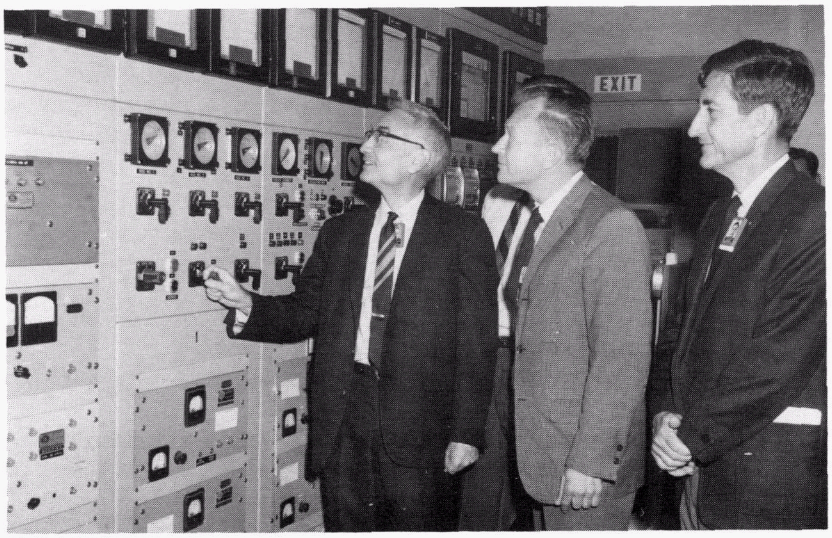


*The Materials Test Reactor (MTR) was designed at ORNL and built at the National Reactor Testing Station in Idaho. Its primary role was to facilitate studies of the influence of high-intensity irradiation on various materials. The reactor was first made critical (reached the "critical condition" wherein a self-sustaining chain reaction state exists) on March 31, 1952. Operation began on May 22, 1952, and the first experimental tests were inserted on August 2, 1952. The MTR was shutdown in 1970 after 17 years and 9 months of operation.*

*The Low-Intensity Test Reactor (LITR) was the original hydraulic-test mockup of the MTR. It was converted to a training reactor for the operating staff of the MTR in 1951 and was subsequently used as a test reactor. The beautiful blue glow of Cerenkov radiation surrounding the fuel of an operating water-cooled and -moderated reactor was seen and photographed for the first time in this reactor. The LITR was shutdown on October 10, 1968.*



*Blue glow of Cerenkov radiation produced in the LITR provides the lighting for this self-portrait of the enriched fuel core of the reactor.*




*Shutdown of LITR. Shown at the controls are, left to right, A. M. Weinberg, S. E. Beall, and J. A. Cox, Operations Division Superintendent.*

# 3. 1951–1961 NONMILITARY AND MILITARY NUCLEAR REACTOR EXPLORATION (Reactor Experimental Engineering and Aircraft Nuclear Propulsion/Aircraft Reactor Engineering/Reactor Projects Divisions)

This chapter describes the two engineering divisions that were formed by dividing the Reactor Technology Division; albeit one had three titles during this period. The division became two when the aircraft nuclear propulsion (ANP) work was moved to a new division in January 1951 with R. C. Briant as director. The Reactor Technology Division became the Reactor Experimental Engineering Division with C. E. Winters as director in July 1951. The primary goal of the latter was to develop aqueous homogeneous reactors for producing power and fuel by breeding through use of the thorium-232/uranium-233 breeding cycle*. The two divisions moved from the Oak Ridge National Laboratory (ORNL) site to the Y-12 Plant in 1951, with the ANP Division occupying Buildings 9704-1 and 9201-3, while the Reactor Experimental Engineering Division moved into 9204-1.†

The decade from 1951 to 1961 was one of many changes in ORNL organizational structure and operation. A major difference was the way in which ORNL assignments were carried out.

By 1951, ORNL technical divisions were mainly organized to work in specialized areas such as chemistry, metallurgy, physics, and solid state physics. Further, ORNL was so well staffed that there was a resident leading expert in virtually any of the fields of science or technology that might be involved in a reactor project. In cases where a resident expert did not have the information needed, that person could quickly make informal inquiries with colleagues in other organizations to ascertain what information might be available.

Projects for development of reactors or power plant systems, on the other hand, required the application of a full range of disciplines in the engineering work necessary for design and construction of components and operating systems. Because avantgarde projects were being addressed, there was need for extensive input from those engaged in the wide variety of research that the project was designed to exploit. Therefore, matrix-type management was introduced during the 1951 to 1961 decade and extensively used.

When a reactor project was centered in the project division with a core team of full-time people, their efforts were supplemented by experts in other divisions. This supplemental effort could be carried out through consultation or by having individuals or whole groups from other divisions join the team on either a short- or long-term basis. Contributors to reactor project divisions included Metallurgy, Chemistry, Chemical Technology, Physics, Solid State Physics, Health Physics, Instrumentation and Controls, Applied Nuclear Physics, Biology, and Isotopes. This highly flexible system worked amazingly well because of the remarkably effective management methods of A. M. Weinberg.

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*Breeding was considered to be of major importance from the outset of nuclear energy development because projections showed that overall use would require more uranium-235 than available.

†The buildings at Y-12 were identified by a unique numbering system. Each set of buildings was numbered in series (e.g., 700, 800, 900 series). Building 9704-1 was an office building in the 9700 series, which includes buildings for other uses as well. The buildings for separation of uranium isotopes, or uranium-235 production buildings, are in the 9200 series. Buildings 9201-1, 9201-2, 9201-3, 9201-4, and 9201-5, (or Alpha 1, Alpha 2, Alpha 3, Alpha 4, and Alpha 5, respectively) are production buildings that housed alpha calutrons, the machines for uranium isotope separation. Buildings 9204-1 and 9204-2 (or Beta 1 and Beta 2, respectively) housed beta cyclotrons, which were the same as the alpha calutrons except for being smaller in size. Therefore, the production buildings occupied by the two ORNL divisions are also called Alpha 3 and Beta 1.

In this section, Reactor Experimental Engineering Division work on the Materials Testing Reactor (MTR) and homogeneous reactor development will be described first. This will be followed by descriptions of projects and programs under the ANP Division and its successors, the Aircraft Reactor Engineering and the Reactor Projects Divisions. Organizational charts for 1954 are provided in Appendix B.

## 3.1 REACTOR EXPERIMENTAL ENGINEERING DIVISION

The Reactor Experimental Engineering Division had three directors before it was joined with the Reactor Projects Division in 1960: C. E. Winters 1951 to 1953, J. A. Lane 1953 to 1958, and R. B. Briggs 1958 to 1960. The Homogeneous Reactor Project director reported to A. M. Weinberg throughout the period of project existence. The directors were J. A. Swartout, 1951 to 1957 (C. E. Winters as Associate Director in 1956), R. B. Briggs (with C. E. Winters as Associate Director), 1957 to mid-1959.

### 3.1.1 Materials Testing and Other Reactors

As stated earlier, design and development work on the MTR continued until 1952. M. M. Mann and R. M. Jones were leaders of the MTR Project during 1950 and 1951.

In addition to direct support work in connection with this reactor, the MTR design was adapted at ORNL to provide a reactor that was inexpensive, unusually safe and stable in operation, and as flexible as possible for teaching and research use. This reactor was designed to operate submerged in a large pool of water that provided shielding as well as cooling and moderation for the neutrons. Hence, it became known as the "swimming pool" reactor.

A reactor of this type was constructed at ORNL in 1950 to study, in bulk, candidate materials for use in irradiation shields. Therefore, the reactor became known as the Bulk Shielding Reactor. The reactor was also used in research activities and to demonstrate a low-cost, versatile tool for use in education by universities. Through continued improvement of this MTR-type reactor design, reasonably standardized swimming pool research reactor designs were established. The ORNL Tower Shielding Facility, developed for shielding research, also incorporates a reactor of modified MTR design; this reactor was built in 1953.

Other applications of modified MTR design were used in the Army Package Power Reactor (APPR) (to be discussed later in this section) and the Oak Ridge Research Reactor (ORR). ORR-type and swimming pool reactors were built in several foreign locations. The first was designed at ORNL for a small nuclear plant that could be installed at remote or relatively inaccessible locations. [The High-Flux Isotope Reactor (HFIR), discussed in the fourth section of this report, is similar to an MTR-type.] Thus, it is clearly evident that the MTR concept had a profound influence on reactor development at ORNL and internationally as well.

### 3.1.2 Homogeneous Reactor

Subsequent to the 1945 examination of aqueous homogeneous reactors, an apparent solution had been found to the need for a suitable material to withstand the temperatures and pressures required to minimize the bubble problem. Zirconium, with a low neutron absorption cross section as well as good strength and corrosion resistance, was the candidate. In addition, considerable experience had been gained in handling radioactive liquids at high temperature and pressure; many components had been tested for high-temperature and -pressure use. Finally, research on uranyl sulfate and water solutions showed promise for their use as fuel solutions in homogeneous reactors. Therefore, the outlook for homogeneous reactor development was good. As stated previously, approval for construction of a homogeneous reactor experiment (HRE) was granted in September 1950.

**HRE**. The HRE (or HRE-1) was to demonstrate the feasibility and operating characteristics of homogeneous reactors. Specific objectives are briefly stated as follows:

1. demonstrate operating feasibility of a circulating fuel reactor at power,
2. study irradiation-induced decomposition and corrosion at power densities on the order of those that might be encountered in a power reactor,
3. demonstrate electric power production, and
4. obtain operating experience in handling a homogeneous reactor at high temperature and pressure.

The core of the HRE was spherical with an 18-in.-diameter, 3/16-in.-thick, stainless-steel container through which 100 gal/min of enriched uranyl sulfate dissolved in distilled water was circulated. The inlet temperature was 410°F, and the temperature rise was 72°F at the design maximum power level of 1 MW. The 482°F fluid from the core was cooled by evaporating water from the shell side of a U-tube heat exchanger, producing ~3000 lb/h of 200-psi steam for generating electricity.

The reflector was a 10-in. layer of deuterium oxide (or heavy water) that surrounded the core vessel. The heavy water was normally pressurized with helium to within ±100 psi of the fuel pressure to minimize stresses in the wall of the spherical fuel container, and the reflector temperature was regulated near 350°F. Both the reflector and the concentric core were contained in an outer, forged-steel, pressure vessel with a 39-in. diameter and a 3-in.-thick wall.

This reactor had a negative temperature coefficient; that is, when power demand was increased, the reactor automatically responded with an increase in temperature rise to produce an equal increase in power. Also, when power demand was decreased, power output was decreased. Thus, the reactor was self-stabilizing.

Before operation, a thorough and rigorous program of testing and inspection was carried out. Major emphasis was placed on inspection and tests to ensure that the system would be leak-tight under full operating pressure. Critical experiments were carried out at low power and high temperature over a 3-month period.

Full-power operation (1-MW heat output at 482°F and 1000 psi) was achieved on February 24, 1953. A sufficient quantity of 200-psi steam was produced to generate 150 kW of electricity. This followed the demonstration of electric power production from nuclear energy in ANL's Experimental Breeder Reactor by only 2 months.

The HRE was subjected to a complete program of tests for determining its characteristics and behavior. These continued successfully through the remainder of 1953 and the early part of 1954. All of the objectives were met with very assuring results. The reactor was dismantled in 1954 to allow a larger homogeneous reactor to be built in its place in keeping with continued development of large-scale power reactors.

Main contributors to the successful outcome for the HRE were the Homogeneous Reactor Project Director J. A. Swartout; the division directors, C. E. Winters and J. A. Lane (see Table A.1); the section leaders in the 1951 to 1954 period; and coworkers as well as supporting personnel from other divisions. Section leaders were R. N. Lyon, C. B. Graham, S. E. Beall, L. R. Quarles, W. R. Gall, E. G. Bohlmann, W. M. Breazeale, R. B. Briggs, H. F. Poppendiek, J. N. Baird, and M. C. Edlund. Supporting personnel included F. L. Culler and F. R. Bruce, Chemical Technology; E. C. Miller, Metallurgy; and S. C. Lind, C. H. Secoy, and H. F. McDuffie, Chemistry.

HRE operation demonstrated the following characteristics for this system:

1. inherent nuclear stability,
2. lack of need for mechanical control rods,

3. direct dependence of reactor power on turbine demand,
4. flexibility and simplicity of fuel handling,
5. the ability to attain and maintain leak-tightness in a small high-pressure reactor system,
6. safe handling of hydrogen and oxygen produced by irradiation decomposition of the water, and
7. the use of copper sulfate as a homogeneous catalyst for recombining these gases as they formed in the fuel.

Expansion of the program was ensured by these results in addition to other encouraging results from concurrent development programs. Development of a large thorium breeder reactor was the ultimate goal.

Because a slurry* of thorium oxide and deuterium oxide was to be used in demonstrating reactor breeding capabilities, studies of slurries were conducted by R. N. Lyon and coworkers, with a difference being that light water was used instead of heavy water. These were out-of-reactor studies addressed to handling and pumping of these slurries and to investigating erosion, corrosion, and caking actions. The test loops used were housed in Building 9204-1.

**Homogeneous Reactor Test (HRT).** The Homogeneous Reactor Test, or HRE-2, was built to take the second step toward a full-scale power station. Located at the site formerly occupied by the HRE, it was an advancement over its predecessor in power, physical size, and quality of construction.

The core was contained in a 32-in.-diameter pear-shaped vessel fabricated of Zircaloy-2, which is a zirconium-tin alloy with a very low neutron absorption cross section. The inlet diffuser section was made from 3/8-in. plate formed into two truncated cones, one having a 30° angle and the second a 90° angle. These were attached to a spherical segment to complete the lower part of the vessel. The top portion was a hemispherical shell with a 5/16-in. wall. The fuel solution of uranium sulfate with copper sulfate in deuterium oxide entered the bottom of the core through a 3-1/2 in. inlet pipe. The inlet temperature was 494°F. Nine perforated plates 1/8 in. thick were arranged to form a diffuser located in the conical sections to distribute the entering fluid. The fuel was circulated at 400 gal/min. When operated at design conditions of 2000-psi pressure and 572°F outlet temperature, the reactor power produced was 5 MW(t).

Because corrosion was found to be less severe at temperatures above 482 to 572°F than at lower temperatures, the HRT was designed to operate within or above that range. The higher operating temperature greatly increased the attractiveness of homogeneous reactors for producing power. The steam was used partially to generate electricity, with the remainder going to an air-cooled steam condenser.

Surrounding the core vessel was a second vessel designed for a 2000-psi operating pressure. Having a 60-in.-inside diameter, it provided an annular space, or blanket thickness allowance, of 14 in. between the two vessels. It was fabricated from two hemispheres made of 4-in. carbon steel clad with a 0.4-in. layer of type 347 stainless steel. Power was to be extracted from both the core and blanket solutions by pumping them through external heat exchangers to produce steam.

Three groups of experiments were planned based on the use of three different blankets: (1) a pure deuterium oxide blanket to demonstrate long-term reliable operation and practical maintenance, (2) a slurry of thorium oxide and deuterium oxide at high concentration to demonstrate breeding of uranium-233, and (3) a low-enrichment but high-concentration uranium sulfate solution in the blanket to demonstrate the production and separation of plutonium-239.

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*Watery mixture of insoluble matter.

The objectives of the HRT were to

1. demonstrate that a homogeneous reactor of moderate size can be operated with the continuity required of a power plant;
2. establish the reliability of fuels, engineering materials, and components with features that can be adapted to full-scale power plants;
3. evaluate equipment modifications that will lead to simplifications and economy;
4. test maintenance procedures and, in particular, maintenance under water; and
5. develop and test methods for the continuous removal of fission and corrosion contaminants for which the HRT was supplied with an integral fuel processing plant.

The design of the HRT was started in January 1954. Construction of the reactor system and the high-pressure system of the associated processing plant was completed in the summer and early fall of 1956. Cleaning, preliminary testing, and nonnuclear operations as well as critical experiments were carried out on a schedule that allowed reactor criticality to be reached on December 27, 1957. First full-power operation occurred in February 1958.

Construction of the HRT was made possible by the development of new or improved equipment for handling uranyl sulfate solution fuel at high temperature and pressure. This equipment was the product of combined efforts by ORNL and industry personnel.

Maintenance of the homogeneous reactor system required considerable development work. Major items were tools and methods for remotely inspecting and repairing highly radioactive components and systems in situ. These were successfully developed and effectively utilized during the operating period of this reactor system.

In the case of the HRT, major contributors include the Homogeneous Reactor Project Directors, division directors, section heads, and coworkers. The division directors were J. A. Lane from 1954 to 1958 and R. B. Briggs from 1958 to 1960 (see Table A.1). Assistant or associate division directors were R. B. Briggs. E. G. Bohlmann, and R. N. Lyon. Section heads in the period from 1954 to 1960 were R. N. Lyon, C. B. Graham, H. F. Poppendiek, S. E. Beall, J. N. Baird, E. G. Bohlmann, R. B. Briggs, M. C. Edlund, W. R. Gall, I. Spiewak, P. R. Kasten, R. B. Korsmeyer, D S. Toomb, J. C. Griess, E. L. Compere, H. C. Savage, and G. H. Jenks.

Remote inspection and maintenance of the HRT were important factors. Work in these areas was done by S. E. Beall, I. Spiewak, W. R. Gall, M. I. Lundin, J. R. McWherter, F. N. Peebles,* and others. Additional support was provided by personnel from other divisions, as was the case for the HRE. The latter included D. E. Ferguson, Chemical Technology; H. F. McDuffie and E. H. Taylor, Chemistry; and M. T. Kelley, Analytical Chemistry.

Power experiments were started during the last week in March and were continued through April 4, 1958, during which time the initial power level of 3 MW(t) was increased to 5 MW(t). Within a short time at 5 MW(t), a leak developed in the core vessel, permitting the fluid to be transferred from the core to the blanket region. Operation was continued for a brief period, and the reactor was shut down.

The reactor was inspected to determine the location and cause of the leak, but it was not possible to positively identify a hole or crack to account for it. Following many unsuccessful attempts, efforts to locate the cause were discontinued, and attention was given to determining ways to resume nuclear operation.

Tests were conducted in which heavy water and uranyl sulfate solutions were circulated in both the core and blanket systems. On the basis of these

*F. N. Peebles became Dean of Engineering at the University of Tennessee in 1968 and held that position until his death in 1980.

tests, it was shown that the reactor could operate satisfactorily with fuel in both the core and blanket regions and that fuel concentrations could be controlled to produce 60% of the power in the core.

Operations were continued on June 4, 1958. After exploring reactor behavior at progressively higher core average temperatures and power outputs, the reactor was operated routinely for 30 d at 536°F and 3.5 MW(t).

Additional attempts were made in July 1958 to conduct satisfactory examinations of the reactor core vessel and the interior of the outer vessel; but, again, it was not possible to identify the region of failure. The reactor was operated intermittently from August 1958 through 1959 at power levels up to 5 MW(t). The performance was free from major mechanical difficulties, and experience with reactor maintenance exceeded expectations.

Operational experience during 1959 drew attention to questions regarding phase stability of uranyl sulfate solution fuels under reactor operating conditions; the use of austenitic stainless steel; and the design, operation, and metallurgy of the Zircaloy-2 core vessel. Indications were that the uranium was separating from the solution and concentrating at spots along the core vessel wall. The local heat generation by this concentrated material was greater than the cooling rate, causing local overheating of the wall and subsequent hole formation.

During 1959 and January 1960, experimental operation of the HRT set an unique record for reactors of all types by operating continuously for 105 d at power levels up to 5 MW(t). The advantage of on-line fuel addition and fission product poison removal associated with the use of the fluid fuel system were clearly demonstrated. Operation was ended on January 22, 1960, because of clear indication that a second hole had developed in the core vessel.

Actions were again taken to permit continued operation. The diffuser plates in the bottom part of the core vessel were removed, and the two holes, which were now exposed and could be located, were plugged. In addition, the direction of fuel flow through the core was reversed so that the solids that entered or that were formed in the core could be flushed out and better cooling of the vessel wall surface would be promoted. The acidity of the fuel was also increased to raise the phase separation temperature. Work on these modifications spanned most of 1960, with the reactor being ready to resume operations in November 1960.

However, Congressional reaction to the homogeneous reactor experience was unfavorable; under strong attack from Congress, the AEC, on December 28, 1960, instructed ORNL to terminate the homogeneous reactor program by July 1, 1961. Research activities were to be closed out as quickly as could be done in a timely manner. In addition, the AEC requested that the HRT be operated at near full power for 2 to 3 months before to final shutdown.

ORNL strongly protested these decisions and advocated that the long-range goal of developing a thorium breeder be vigorously pursued. The AEC agreed to continuing support only for thorium breeder technology. Therefore, R&D work on thoria slurries and other aspects of thorium breeder reactor development were pursued under a new thorium utilization program.

The HRT was operated at full power, beginning in January and terminating on April 28, 1961, when it appeared that a plug in the core vessel wall had failed. The reactor and systems were dismantled, and final reports prepared to close out the program in 1961.

## 3.2 ANP/ AIRCRAFT REACTOR ENGINEERING DIVISION

A. P. Fraas captured the mood of the time with the following observation regarding participation in the ANP Division activities. "A tremendously important and absolutely unforgettable element of the ANP Program was a marvelous esprit de corps. The sense of excitement, urgency, and degree of dedication that prevailed can't be conveyed to one who didn't experience it, but those who worked on the program count it as one of the greatest events of their lives."

To underscore the enthusiasm and excitement of the staff, A. P. Fraas recalls that for one physicist on the project, C. B. Ellis, this enthusiasm went beyond ordinary limits. Ellis had been the first recruit to the ANP Project at ORNL and A. M. Weinberg's principal lieutenant in making detailed arrangements for setting up the Technical Advisory Board (TAB) and recruiting the first dozen people for the project. Someone had proposed solving the shield weight problem* by using an unmanned nuclear airplane as a tug to tow a manned aircraft at the end of a long cable. Ellis, carried away with enthusiasm for this nifty solution, made a brightly colored banner with the words TUG-TOW in letters a foot high; one morning before anyone else had come to work, he strung it across the hall just above head-height in front of his office where everyone coming into the building would be sure to start the day right by getting the new message! R. C. Briant, now director of the project, was fit to be tied! (After a few more remarkable bursts of enthusiasm, Ellis left to join more appreciative associates.)

### 3.2.1 Aircraft Reactor Experiment (ARE)

The ARE was the outgrowth of the TAB recommendation that prime emphasis be on a high-temperature, liquid-cooled reactor. It was actually a high-temperature liquid-fueled reactor in which beryllium oxide (BeO) blocks were used for both moderator and reflector. The fuel, developed by chemists assigned to the project (W. R. Grimes and coworkers), was a eutectic melt of sodium fluoride and zirconium fluoride with a few percent of uranium fluoride. A complicating factor was the high fuel melting temperature, which was about 1000°F.

The development and use of liquid fuel was done at the insistence of R. C. Briant, Director of the ANP Division and of the ANP Project. (He reported to A. M. Weinberg in the latter role.) His objective was to get rid of the "expensive, lacy, fine structure" of thousands of fuel pins only a few millimeters in diameter that would otherwise be required.

The BeO blocks used in the moderator and reflector regions were hexagonal in cross section, 3.73 in. across flats and 6 in. high. They were stacked to form a circular cylinder 48.32 in. in diameter and 35.64 in. high; the active core lattice was 33.3 in. in diameter. Overall size of the structure was the same as had been estimated for a full-scale 100-MW(t) reactor to be used in a nuclear airplane. However, for the initial experiment, the design power selected was 1.5 MW(t).

BeO blocks in the core region had central holes 1.26 in. in diameter to allow passage of the 1.235-in.-outside-diameter Inconel fuel tubes. There were six parallel tubes; each was wound into serpentine coils, passing through the core lattice 11 times.

The reactor was enclosed in a 2-in.-thick Inconel vessel. Liquid sodium was used for cooling the reflector-moderator and other components within the vessel. To promote cooling of the reflector blocks, 0.49-in.-inside-diameter coolant tubes were installed through the centers of the blocks in each column.

The liquid fuel was pumped through the reactor by means of a centrifugal pump. It was routed from the reactor through external helium-cooled heat exchangers and returned to the reactor. Recirculating high-velocity helium for cooling the fuel, in turn, flowed through water-cooled heat exchangers. This arrangement was used to prevent

*Shielding needed for containing or minimizing the escape of irradiation from the reactor presented a major problem because of the weight involved. Therefore, means for minimizing this weight and retaining necessary shielding were widely examined.

mixing of fuel and water should there be a fuel leak. Heat picked up by the sodium was also transferred to water through an intermediate helium circuit.

The exit temperature of the fuel from the reactor was about 1500°F, and the inlet temperature was about 1200°F. The corresponding sodium outlet and inlet temperatures were about 1240 and 1100°F, respectively.

The reactor was inherently self-controlling in the same way that the aqueous homogeneous reactor was self-controlling. Therefore the only control rods needed were a stainless steel regulating rod and three boron carbide safety rods.

A new test building was constructed close to the aqueous homogeneous reactor building that, at that time, was also under construction. Installation of ARE components was started in 1953 and completed about mid-1954. Operation of the reactor took place in October and November 1954.

The reactor was operated at full design power for ~100 h, as planned. Operation was completely satisfactory and provided convincing demonstration of the feasibility of a high-temperature fluid-fuel reactor.

Although the reactor was designed to operate at 1.5 MW(t), it actually ran at a peak output of 2.5 MW(t). A total of about 90 MW-h of high-power operation was accomplished before scheduled shutdown and dismantling took place. A simple way to illustrate the magnitude of the engineering achievement reflected in successful operation of the ARE is to point out that, while the reactor was in operation, every part of the fuel system was literally red hot, including the reactor core, piping, pumps, valves, heat exchangers, and all other components.

Those who participated in major aspects of the ARE Project were as follows. R. W. Schroeder, aided by G. A. Cristy and L. F. Hemphill, was in charge of design work; W. K. Ergen was the resident physicist; H. W. Savage was in charge of experimental work, including component testing; R. G. Affel addressed instruments and controls; E. S. Bettis and J. L. Meem were in charge of reactor operations; and construction and maintenance activities were carried out under the direction of B. H. Webster and J. C. Packard. Metallurgical support was ably provided by metallurgists under W. D. Manly, while W. R. Grimes and coworkers addressed chemical problems.

Successful operation of the ARE represented a number of major achievements in diverse R&D fields. For example, the development of a molten salt fuel mixture that was a chemically stable, noncorrosive, fluid over the desired range of temperature, and satisfactory in its heat transfer properties, was an outstanding accomplishment. The molten salt fuel had an attractive advantage over the aqueous fuel of the homogeneous reactor because it could be used in a reactor system operating with little increase over normal atmospheric pressure. The development of materials that would resist corrosion by red-hot fluoride salt mixtures, and the development of methods for fabricating these materials into a reactor system that would operate reliably at 1500°F were metallurgical achievements of great importance. The design of a reactor of this comparatively new type to operate under these heretofore unheard of conditions and the development of components to go into the reactor system were engineering achievements of unparalleled difficulty.

In operation, the ARE demonstrated again the advantageous features of fluid fuel reactors, including excellent nuclear stability, strong coupling between power demand and power level, and ease of operation and controllability. It is not an exaggeration to say that the ARE was the most advanced reactor type that ORNL had developed. Consequently, its successful operation represented perhaps the greatest achievement of the combined R&D staffs of ORNL.

Tragically, amidst the successful development of his concept for a molten-fluoride-fueled reactor, R. C. Briant died in 1954. It was a blow to the project, but fortunately, W. H. Jordan, who had been a major guiding force in the ANP reactor design effort, became Director of the ANP Project. A few months later, S. J. Cromer, who had been in charge of gaseous diffusion plant design and expansion work at the K-25 Site, became Director of the ANP Division and Codirector of the ANP Project with W. H. Jordan. The Project was under A. M. Weinberg, ORNL Research Director.

Inasmuch as the gaseous diffusion plant expansion was being completed, Cromer soon brought many other engineers (including M. Bender, W. F. Boudreau, R. S. Carlsmith, J. W. Michel, G. Samuels, and D. B. Trauger) from K-25 into the division to staff the expanding ANP effort. Lastly, the name of the organization was changed to Aircraft Reactor Engineering Division.

### 3.2.2 Aircraft Reactor Test (ART)

When compatibility of the molten fluoride with Inconel was established, an intensive effort was directed toward the design of a full-scale reactor that would exploit the potential use of this fluid fuel to best advantage. This would be done simultaneously with ARE activities.

A reactor system for aircraft propulsion must meet demanding requirements regarding power plant size, weight, and performance. In addition, irradiation shielding effectiveness requirements along with weight limitations must be met. All must be appropriately factored into the design of a reactor system that will support aircraft performance within the envelope of need. As to need, design studies indicated that nuclear power plants capable of producing 100 to 300 MW(t) would be required for intended missions. Therefore, a 60-MW(t) ART was a logical intermediate step and was selected because this power level was approximately that needed for an investigation of engineering problems associated with reactors required for high-altitude supersonic strategic bombers.

The concept of a reflector-moderated, circulating-fuel reactor and shield combination was defined by June of 1953. This concept was examined by experts on shielding who concluded that the associated shield designs were sound and uncertainties in shield weights were not large.

Conceptualization and initial design of the ART were done primarily by A. P. Fraas, with input from C. B. Mills and others. The reactor chosen was contained in a 1-in.-thick Inconel pressure vessel with an 55.62-in. outside diameter. This pressure vessel contained a fuel mixture of sodium, zirconium, and uranium fluorides (the same fuel as used in the ARE) and housed the beryllium moderator-reflector, heat exchanger bundles where heat was removed by liquid sodium-potassium (NaK) and a sodium-to-NaK cooling system for the moderator-reflector.* Fuel and NaK pumps were mounted in the upper head of the assembly. The NaK from fuel-to-NaK heat exchanger bundles was to be pumped to external heat exchangers for heat removal as was the NaK from the sodium-to-NaK heat exchanger. The fuel temperature at the core outlet was to be 1600°F. This compact, high-power-density design became known as the "Fireball."

The successful test of the ARE focused increased effort on the ART. Pratt and Whitney Aircraft Division of United Aircraft Corporation (P&W) decided to shift its effort from a sodium-cooled, solid-fuel reactor to the design of a full-scale power plant employing a 300-MW(t), fluoride-fuel reactor to be developed in close cooperation with ORNL. Also, the USAF urged that molten-salt reactor (MSR) development be accomplished at the earliest possible time. To facilitate both ORNL and P&W programs, arrangements were made to transfer about 40 individuals from P&W to ORNL

*In actual aircraft engine applications, heat exchangers for cooling the NaK outside the reactor pressure vessel would replace the jet engine combustion sections.

for 2 or 3 years. This was partly to aid ORNL and partly for the P&W personnel to gain detailed experience in designing, constructing, and operating both vital component tests and the prototype reactor.

The project leaders and their coworkers made major contributions to the success achieved in the case of the ART. In addition to A. P. Fraas, head of the power plant engineering team, the project leaders included the following. W. K. Ergen provided physics support until April 1955 when A. M. Perry assumed this role; H. W. Savage conducted experimental work, including component tests; H. C. Gray, a P&W employee, was in charge of the design group; H. W. Hoffman was in charge of heat transfer and physical properties work; E. S. Bettis had responsibility for construction activities; and W. G. Piper was in charge of reactor test facility modification and construction. E. P. Blizard and coworkers provided support in the shielding design area, W. D. Manly was in charge of metallurgical work, and chemistry-related activities were carried out under W. R. Grimes.

Comprehensive tests were conducted to resolve important problems with components. However, it was recognized that major uncertainties could be resolved only by building and operating the complete reactor and pressure vessel unit at full temperature in a non-nuclear test and subjecting it to severe thermal stresses. Hence, an engineering test unit (ETU), which was essentially a non-nuclear replica of the ART, was designed, and construction was initiated on a schedule slightly ahead of the ART.

An important basic requirement imposed on the component tests was that each component operate for at least 1000 h to ensure requisite reliability as well as give useful life in field operations. At that time, the service life of a jet engine was <1000 h between overhauls.

While the reactor design, component testing, and other activities were being carried out, the ARE was dismantled and removed from the test building. The ARE building was then enlarged and modified for the installation of the new reactor system.

By the summer of 1957, all tests of individual components had been carried to the point where satisfactory results had been obtained, and many parts of the ETU had been fabricated. Prototype fuel and sodium pumps were running smoothly on endurance tests that would extend to over 20,000 h for each.

At this time, cancellation of the entire ANP Program was announced in Washington. The national ANP Program had come under fire because of high cost, and changing military requirements made the achievement of nuclear-powered aircraft less important as a national goal. In fact, the primary reason for termination was that satisfactory performance of ballistic missiles had been achieved in 1956, demonstrating that the key problems of reentry and guidance were solved. Thus, availability of the nuclear airplane was not a vital military requirement. In addition, the hazards associated with a crash had been a serious shortcoming of the concept from its inception. While possibly acceptable if there were no other way to accomplish the strategic mission, the guided missile now supplied a more attractive and much less expensive alternative.

Although ORNL agreed with and accepted the cancellation decision, the request was made to continue the ETU and ART projects to completion because they would be important steps toward development of high-temperature reactors for marine and electric utility service. AEC denied this request but agreed to initiation of a small project to develop an MSR for electric utility service and permitted continuation of some of the test work directly applicable to the civilian MSR. It was apparent that the adaptation of this type of reactor to serve as a central station power plant would involve significantly less technological problems. Further, AEC insisted that the bulk of the personnel working on the ART project was needed to start work on a new gas-cooled reactor (GCR) project that had just been mandated by Congress.

The design of the ART was completed and shelved. The enlarged and modified ARE building in which the ART was to be installed was placed in standby for possible use to house a future reactor experiment, and the orders for ART components and other material that were not delivered were cancelled.

Both GE and P&W succeeded in obtaining continued funding, and a small effort continued at ORNL to provide support for the main line of attack being pursued by the two. P&W personnel who were engaged in project work at ORNL were reassigned to work at P&W facilities. However, many elected either to seek employment with ORNL or to return at a later time to join the ORNL staff.

The national ANP Program was terminated on June 30, 1961. ORNL R&D efforts in this field were shifted to the space power program and to high-temperature materials work.

With the cancellation of work on the ETU and ART, S. J. Cromer left ORNL for an assignment with corporate headquarters of Union Carbide Corporation in New York. W. H. Jordan, who was Director of the ANP Project, assumed the position of director of the division.

The ARE Division activities would be quickly redirected to address gas-cooled, civilian, power reactors in keeping with the mandate from the Congress. In September 1957, a study of this reactor type was assigned to ORNL for immediate action. R. A. Charpie, Assistant Director of ORNL, directed the study.

## 3.3 REACTOR PROJECTS DIVISION

In 1958, the division responsibilities were expanded to include the Army Package Power, Maritime Ship, and MSR Programs in addition to a Gas-Cooled Reactor Program (GCRP). The division title was appropriately changed to Reactor Projects Division.

The abrupt phase-out of ANP work at ORNL brought a number of changes beyond the change in division leadership. Personnel were reassigned, and a search for new projects began. Because GCR work was seen as holding major promise, work in this area was vigorously pursued.

R. V. Mehgreblian, who was in charge of applied mechanics and stress analysis work under A. P. Fraas, was chosen to aid in directing GCR work for a period in 1958. Also, in 1958, A. L. Boch, A. P. Fraas, and H. G. MacPherson were appointed Associate Division Directors. A J. Miller became Assistant Director.

R. A. Charpie became both Division Director and Head of the GCRP in 1959. The Reactor Projects and the Reactor Experimental Engineering Divisions were combined in late 1960; R. A. Charpie was the first director of the combination, which was titled Reactor Division.

### 3.3.1 GCRs

The mandate from the Congress for work on GCRs was precipitated by at least two factors. The British were enjoying enviable success with GCRs, and a GCRP did not exist in the United States. Acting on this mandate in September 1957, AEC undertook, as a part of its Reactor Development Program, a serious study of GCRs for power production. The purpose of this study was to present to the Congress a specific set of conclusions concerning the possible role of GCRs in the United States together with a set of recommendations that would constitute a national program on GCR development.

Individual studies by Kaiser Engineers and ORNL were commissioned. AEC requested that ORNL make the results of its study available on April 1, 1958, with an early construction date in mind. Generally speaking, gas-cooled systems had received hardly more than casual attention because early studies seemed to indicate that it is difficult to achieve sufficiently high power densities in these reactors to make them economically attractive. This notion had remained firmly implanted in American nuclear energy thinking.

The ORNL portion of the study program was to consist of a design study of a graphite-moderated, enriched-uranium-fueled GCR, together with identification of R&D work required for natural- and enriched-uranium-fueled GCRs. As mentioned, R. A. Charpie directed the study.

The reactor system selected for study was for base-load operation, with provision for load following ability. Gross thermal output of the reactor was 687 MW; the net electrical output of the power plant was 225 MW. The reactor was helium-cooled, with gas outlet and inlet temperatures of 1000 and 460°F, respectively. The working pressure was 300 psia.

The reactor system was titled ORNL Gas-Cooled Reactor-2 (or GCR-2). The primary differences between the GCR-2 and the British Calder Hall (or Magnox GCR) were the use of stainless-steel clad fuel elements, enriched uranium oxide fuel, and helium gas coolant in the first vs magnox (a magnesium material) fuel cladding, carbon dioxide coolant, and nonenriched fuel in the second.

These principal conclusions were from the six-volume report on the ORNL study:

1. Graphite-moderated GCRs have good future prospects for application in the United States.
2. Enriched uranium-fueled GCRs will produce power more cheaply than natural-uranium-fueled reactors, as employed by the British.
3. GCRs are technologically and economically competitive with pressurized-water reactors (PWRs) for power production.
4. Helium-cooled, graphite-moderated reactors utilizing 2% enriched uranium oxide fuel elements clad with stainless steel and having a maximum gas temperature of 1000°F represent a good starting configuration comparable with current technology.

Reconciliation of results from the ORNL and Kaiser Engineers studies brought the overall conclusions into essential agreement with those from the ORNL study.

A. M. Perry considers his involvement in the GCR-2 studies as the most exciting and productive era in his career. Within 2 weeks of closeout of the ART, GCR redeployment was completed, and the staff was up to speed for conducting the necessary work within about a month, despite the fact that the staff was inexperienced in civilian nuclear power plant design. From September 1957 to January 1958, the subject was researched and addressed, and a definitive report to guide future U.S. reactor development was produced—an astonishing feat.

Based on the results obtained, a fund for prototype reactor plant design and construction was established. Kaiser Engineers was selected to design a prototype reactor to be built at the Nuclear Reactor Testing Station (NTRS) in Idaho.

Kaiser Engineers had worked with ACF Industries as a nuclear subcontractor during the study phase. Therefore, Kaiser Engineers teamed up with the same group for the prototype design. Before teaming for the second time, the ACF Industries organizational unit had been purchased by Allis Chalmers Manufacturing Company. The nuclear work was therefore carried out by Allis Chalmers in Washington, D.C., while the remainder was done by Kaiser Engineers in Oakland, California.

The demonstration prototype reactor plant was to have an electrical output of 30 MW. Idaho Operations Office of AEC was to manage the project. ORNL was given design review responsibility, which entailed a number of visits by the ORNL review team to the Kaiser Engineering offices in Oakland during 1958 and 1959.

F. H. Neill was initially given responsibility for oversight of ORNL work in connection with the demonstration prototype reactor. This work, including the design review activity, was assigned to M. Bender when design review activities began.

During the initial design phase, it was recognized that the NTRS electrical power grid was too small to accommodate an abrupt loss of 30-MW(e) input, as was likely to occur with an experimental reactor plant. On this basis, the plant site was changed to Oak Ridge, where the plant could be tied into the Tennessee Valley Authority (TVA) grid, a grid large enough to be essentially unaffected by an abrupt 30-MW(e) loss of power input.

Having made this change, Oak Ridge Operations Office of AEC (ORO) became project manager, with L. H. Jackson in charge. In April 1959, design work on the Experimental Gas-Cooled Reactor (EGCR) under ORO began. The responsibility for the design and construction of the facility remained with Kaiser Engineers and Allis Chalmers Manufacturing Company. ORNL was assigned responsibility for the detailed design of the reactor fuel and reactor control rods as well as development work on reactor components. Union Carbide Nuclear Company was assigned responsibility for procurement of the reactor fuel, control rods, and control rod drives. ORNL was to serve as technical advisor to AEC in continuing review of the detailed design. At a later date, Union Carbide Nuclear Company would also be assigned the responsibility for the design of an emergency core cooling system and for procurement of the equipment for this system. The H. K. Ferguson Company was selected as construction contractor, and TVA was selected to operate the facility for AEC. Construction of the EGCR was initiated in August 1959.

The reactor plant was to be located on the Clinch River at Gallaher Bend in the Oak Ridge Reservation. Reactor thermal power was to be 84.3 MW, with a plant net electrical output of 29.5 MW. The reactor outlet temperature of the helium coolant was 1040°F; the corresponding pressure was 314 psia. The inlet temperature was about 510°F. The moderator-reflector material was graphite, and enriched uranium in the form of uranium oxide pellets was used for fuel.

The principal function of the EGCR was to demonstrate the power production capability and to obtain information that could be applied to the design and operation of future GCRs. Additionally, provisions were made for installation of experimental loops to enable the reactor to be used as an experimental facility at a later time. These loops were a province of ORNL.

M. Bender was assigned responsibility for the EGCR Project at ORNL. In addition to participation in the technical adviser role and responsibility for detailed design of the reactor fuel and control rods, the Reactor Projects Division carried out direct support work. Fuel design work was spearheaded by G. Samuels, and control rod design was led by J. W. Michel. As a part of direct studies, for example, experimental stress analysis work on the pressure vessel and internals was done, and significant input was given on design and analysis of graphite core components. ORNL and the division also became involved in reactor physics and design studies; experimental investigations of heat transfer and fluid flow; materials (including fuel, graphite, and structural metals); out-of-reactor testing of components; and in-reactor testing of fuels. Finally, ORNL accepted responsibility for continuing advanced studies aimed toward developing a reactor of improved performance. EGCR Project leaders are listed in Table 3.1.

The EGCR design was among the first, if not the first, to include seismic considerations. Seismic considerations, up to that time, were addressed to site selection rather than to reactor system design.

Because the yoke of secrecy was virtually removed, GCR work brought with it exchanges with representatives working on similar or related studies in other countries as well as within the

Table 3.1. EGCR Project Personnel (Reactor Projects/Reactor Division) 1960–1961

| Name | Responsibility |
|---|---|
| A. M. Perry<br>C. A. Preskitt | **REACTOR PHYSICS** |
| | **REACTOR DESIGN STUDIES** |
| G. Samuels | Thermal Analysis of Core Components |
| J. W. Michel<br>G. Samuels | Control Rod |
| W. L. Greenstreet | Structural Investigations |
| W. B. Cottrell<br>M. H. Fontana | Hazards Evaluation |
| A. B. Meservey | Decontamination of EGCR Components |
| H. W. Hoffman<br>J. W. Wantland | Experimental Investigation of Heat Transfer and Fluid Flow |
| J. H. Coobs[a] | **MATERIALS RESEARCH AND TESTING** |
| D. B. Trauger<br>O. Sisman[b] | **IN-PILE TESTING OF COMPONENTS AND MATERIALS** |
| D. B. Trauger | Fuel Element Irradiation Program |
| R. G. Berggren[c] | Irradiation Effects on Structural Materials |
| Members of Metallurgy and other divisions | **OUT-OF-PILE TESTING OF MATERIALS** |
| H. W. Savage | **DEVELOPMENT OF TEST LOOPS AND COMPONENTS** |
| F. H. Neill | EGCR In-Pile Loops |
| R. E. MacPherson | EGCR Component Tests |
| J. Zasler | GCR–ORR Loop Design and Construction |
| W. F. Boudreau | Special Compressors |

[a]Metallurgy Division Coordinator.
[b]Reactor Chemistry Division.
[c]Metallurgy Division.

United States. Initially, the British and French were the primary foreign exchange participants, but this also changed. In this country, the exchanges included those with personnel from government agencies, private companies, colleges, and universities. Thus, broader perspectives and wider recognition accrued to those involved in the program.

During the design phase of the demonstration prototype plant, General Atomic, in San Diego, designed a demonstration prototype GCR and joined with a group of utility companies led by Philadelphia Electric Company to build a nuclear plant, the Peach Bottom Plant at Lancaster, Pennsylvania. On this basis, General Atomic and the Philadelphia Electric Company group were able to obtain sufficient support from the AEC fund for prototype plant design and construction to proceed. Hence, two GCR projects were being carried out in parallel under AEC funding, with the Peach Bottom Plant being a strong competitor to the EGCR. The Peach Bottom reactor output was 115 MW(t) or 40 MW(e). It reached criticality in 1966 but was permanently shut down in 1974.

### 3.3.2 Small Reactors—APPR and Maritime Ship Reactor

#### 3.3.2.1 APPRs

In 1952, the U. S. Army was assigned responsibility for the development of land-based nuclear power plants required by the military services for heat and power at remote locations. The Army assigned this responsibility to the Corps of Engineers, and an Army Nuclear Power Program was established as a joint program of the Army and AEC. An Army Reactors Branch (ARB) was subsequently organized in the Division of Reactor Development of AEC in December 1952, and the APPR Project was initiated in the Electromagnetic Division (R. S. Livingston, Director) at ORNL early in 1953. A. L. Boch was the project leader. Others associated with the project included F. H. Neill, H. C. McCurdy, and A. M. Perry.

The objective was to exploit the compactness of a nuclear power unit by developing a small power plant that could be installed at remote or relatively inaccessible locations. These locations were to be those where nuclear power costs would be competitive with conventional power costs in the area and where the fuel, because of its competitiveness and potentially long life, would have logistic advantages over other fuels.

Thus, the first task addressed by ORNL was to design, in a short time, a practical nuclear plant that could furnish electrical power in certain isolated localities at a cost that could be competitive with the cost of power produced by existing power generating facilities in similar locations. R. B. Briggs was obtained on loan from the Reactor Experimental Engineering Division to aid in the conceptual design; W. R. Gall also made significant design contributions.

Based on the requirements given, a PWR was selected. The reactor was of the MTR type, being water-cooled and -moderated and utilizing fuel similar in design to the fuel elements employed in the MTR. Except, in this case, the fuel was for relatively high-temperature use. It was, therefore, made up of uranium oxide pellets uniformly dispersed and imbedded in a matrix of stainless steel and clad with stainless steel.

Water was circulated through the reactor at 4000 gal/min by one pump, with a duplicate pump in reserve. The water system was maintained at 1200 psi to preclude boiling; reactor inlet temperature was ~430°F, while the outlet temperature was 450°F (when operating at full load). After leaving the reactor, the water was circulated through a steam generator to produce steam for driving a turbine generator for producing ~1.9 MW of electricity.

The conceptual design of this 10-MW(t) reactor plant, which was designated APPR-1, was completed in the summer of 1954.

Late in 1954, American Locomotive Company, which subsequently became ALCO Products, Inc., was selected to build the APPR-1, based on the ORNL design, at Ft. Belvoir, Virginia, as the first Army reactor. The APPR-1, since renamed the SM-1 to correspond with current Army nomenclature, was completed and taken to criticality on April 8, 1957.

ORNL contributions to the SM-1 project included the following, in addition to the conceptual design of the reactor and power plant: (1) assistance in technical aspects of bid preparation and evaluation; (2) design and performance of critical experiments to determine nuclear characteristics; (3) development and fabrication of the reactor core, which was the first of its kind; (4) developing and irradiation testing of fuel and control materials; and (5) technical review during design and construction phases. To carry out these activities, support was obtained from other ORNL divisions, including Metallurgy, Chemistry, and Physics. Materials were developed, and the core was fabricated in the Metallurgy Division under the technical direction of J. E. Cunningham. C. F. Baes, Jr., and A. D. Callahan were primary contributors in the Chemistry and Physics Divisions, respectively. In all, the APPR group acted as the technical arm for the ARB in addition to participating directly in the work involved.

The SM-1 reactor plant was completed in less time than scheduled (31 vs 36 months), and the performance met or exceeded expectations. It was the first to be built on a fixed-price contract. In addition, the reactor was the first to use a burnable poison in the fuel to maintain constant reactivity with time for a designated period. The plant was shut down in the 1970s.

The remainder of the ORNL effort was directed largely toward support of ARB activities dealing with developments other than the PWR. Specifically, support was given to the mobile GCR program, and several Oak Ridge School of Reactor Technology design studies were sponsored; they provided initial design concepts for reactor development programs.

The APPR Project was transferred to the Reactor Projects Division in 1958, and A. L. Boch became Associate Director of that division. H. C. McCurdy was appointed to head the APPR Project and a new Maritime Ship Reactor Project. Later, L. D. Shaffer was in charge of the APPR Project for a short period until his death in an airliner crash on returning to Knoxville from Washington, D.C.

In the spring of 1958, the decision was made to remove ORNL from direct support roles, as exemplified by work on the SM-1 reactor plant. A reduced level of effort was established wherein support of the ARB was mainly devoted to R&D work on metallurgical aspects of PWR systems, with occasional consultation in other areas.

Work was subsequently done in connection with both stationary and portable, or mobile, plants.* These included descendants of the SM-1; Martin-Marietta Company's version of the same general type as the SM-1, but differing in detail (employing, for example, tubular instead of flat plate fuel elements); and a mobile, GCR, closed-cycle, gas-turbine system championed by Aerojet-General Nucleonics. ORNL involvement embraced reviews of detailed designs and specifications for the nuclear systems and some supporting development work associated with the reactor core and control rods as well as with various aspects involving metallurgy and chemistry.

Design as well as specification and fabrication of replacement APPR cores were closely followed by ORNL. In addition, the cores were inspected immediately after unloading from shipment. Post-irradiation examination of the fuel and control

*P denotes a portable and S denotes a stationary reactor system.

rods was carried out until the ORNL project was terminated in about 1966.

Example deployments of APPRs are as follows. The PM-2A, a transportable version of the SM-1, was located in snow tunnels at Camp Century, which is on the Greenland Icecap 900 miles from the North Pole.

The PM-1 reactor plant, developed by Martin-Marietta Company, was installed at the Air Defense Command atop Warren Peak, 7 miles from Sundance, Wyoming. It was first made critical on February 5, 1962.

Another Martin-Marietta Company reactor plant, the PM-3A, achieved criticality only 8 d later; it was the first reactor plant to operate on the Antarctic Continent. The PM-3A plant supplied part of the power requirements for the Naval Air Facility at McMurdo Sound, which was the main support base for U.S. operations in Antarctica. A result was a measurable increase in quality of life for those stationed there.

#### 3.3.2.2 Maritime Ship Reactor

The Maritime Ship Reactor Program at ORNL was carried out in support of the N.S. Savannah project, which was a joint responsibility of AEC and the Maritime Administration. This national project resulted from a 1955 proposal by President Eisenhower for the United States to build a nuclear-powered ship to demonstrate peaceful uses of atomic energy to the world. It was to be a part of the Atoms For Peace Program as were the Geneva Conferences, held in 1955 and 1958.

In July 1956, Congress authorized construction, and on October 15, 1956, the President directed AEC and the Maritime Administration to proceed. AEC was made responsible for providing the power plant, and the Maritime Administration was responsible for providing the ship and other equipment, for training crews, for providing fuel-handling facilities, and for operation of the ship. A joint group was therefore established to carry out the Savannah project and to plan further development of nuclear-powered surface ships. Following the lead set by the APPR project, a Maritime Reactors Branch (MRB) was established under the Division of Reactor Development of AEC.

The Savannah was a single-crew, passenger-cargo ship capable of carrying 10,000 tons of general cargo, 60 to 100 passengers, and a crew of 109. She had an overall length of 595 ft, a beam of 78 ft, and a full-load displacement of 22,000 tons. With a shaft horsepower of 20,000 (22,000 maximum), she had a normal speed of 21 knots. Power was furnished by a low-enrichment 69-MW(t) PWR.

The reactor and propulsion equipment were designed and furnished by the Babcock and Wilcox Company; George Sharp, Inc., was the ship architect; and New York Shipbuilding Corporation was responsible for construction of the ship and test operations. States Marine Lines was contracted to operate the ship for the Maritime Administration. Fixed-price contracts were placed with the first three.

After the contracts were signed, MRB proposed to make use of ORNL in a role similar to that established by the ARB for the APPR Project. Major items were design and development review and inspection assistance as well as participation in development tasks. This proposal was accepted by ORNL, and, in September 1957, ORNL began a program of technical support to the MRB in the development of nuclear-powered ships. This support was coordinated by the APPR group of the Reactor Projects Division with general guidance from the Maritime Steering Committee composed of senior members of the ORNL staff.

The APPR group under H. C. McCurdy reviewed reactor designs, specifications, and development programs being carried out by AEC contractors and advised MRB on technical matters. The assistance of numerous specialists throughout ORNL was enlisted for this review and advisory service. In addition, supplemental studies were undertaken as the need arose. An example is hazards review work for the Savannah done by W. B. Cottrell and coworkers. Also, a pressurized-water loop for irradiation testing of fuel was designed, installed in the ORR, and operated by the Irradiation Engineering Group under D. B. Trauger.

Inspection engineering activities, including witnessing inspections and tests during fabrication of some components relating to the reactor and associated equipment, were carried out by the Inspection Engineering Department under E. C. Miller. T. J. Burnett of the Health Physics Division coordinated activities to develop bioassay methods for determining internal irradiation exposure of Savannah personnel and prepared a health physics manual; waste disposal was addressed by W. J. Neill and others. Irradiation shield survey work was done by T. V. Blosser and coworkers in the Neutron Physics Division. Work on gas filters for the reactor compartment emergency and normal ventilation systems were led by W. E. Browning of the Reactor Chemistry Division. Finally, reactor controls were investigated by E. R. Mann of the Instrumentation and Controls Division.

The first full-power operation of the Savannah reactor occurred in April 1962. The first port visited was Savannah, Georgia, in August 1962. She subsequently visited ten ports (Norfolk, Seattle, San Francisco, Long Beach, Los Angeles, Honolulu, Portland, San Diego, Balboa*, and, finally, Galveston, the port used for maintenance and refueling) between August 1962 and February 5, 1963. About 30,000 nautical miles were covered. During this period, she had ~340,000 visitors, with the maximum for 1 d being greater than 12,000. By March 1965, she had visited 55 foreign and domestic ports and been viewed by 1,500,000 people.

The passenger section was converted for cargo use in 1965, and, for the next 5 years, the Savannah carried cargo on a commercial basis. She traveled to 77 ports of call during this period; of these, 66 were in 25 foreign countries.

Although the support program essentially ended in 1964, ORNL (S. I. Kaplan and O. Klepper) continued to aid in obtaining port clearances throughout the total span of operation. The ship was decommissioned in 1971, and the reactor plant was removed. She was presented to Savannah, Georgia, in 1972 and is now displayed at Patriots Point Naval and Maritime Museum, Charleston Harbor, South Carolina.

### 3.3.3 MSR Program

During the development of ANP technology for MSRs, it became apparent that this type reactor offered inherent advantages for electric power production. Following the agreement reached with AEC for pursuit of civilian power applications, a civilian MSR program was established in 1958 under the direction of H. G. MacPherson. At the same time, the former ANP molten salt groups were able to provide personnel for a smaller scale effort to adapt ANP technology to achieve a civilian power reactor.

MSRs were recognized to combine, almost uniquely, the advantages of very high temperature, wide solubility limits of the fuel, and low pressure in a liquid system. Because of the low working pressure, the mechanical parts of the system were relatively uncomplicated. It was hoped that this basic simplicity would offset the cost of required heating (to about 1000°F) to melt the fuel and remote maintenance equipment, making capital costs nearly equal to those of other power reactors when compared on a heat generation basis. This coupled with higher thermal efficiency due to higher temperature operation would then give appreciable advantage in capital charges. Lowered fuel costs also were believed to accrue to these liquid-fuel reactors. These basic considerations justified pursuit of the development effort.

*Panama Canal Zone.

ORNL, by this time, had developed a nickel-molybdenum alloy, called INOR-8,* for containment of molten-fluoride salts at high temperature. It was resistant to oxidation and corrosion, possessed good welding properties, and had good high-temperature strength. Several firms were willing to supply the material. In addition, phase studies of molten-salt mixtures showed suitability both for fuel and breeding (blanket) material; a basis for continuous removal of fission product gases (off-gas) from MSR was shown, and studies established that plutonium fuel can be used in a MSR.

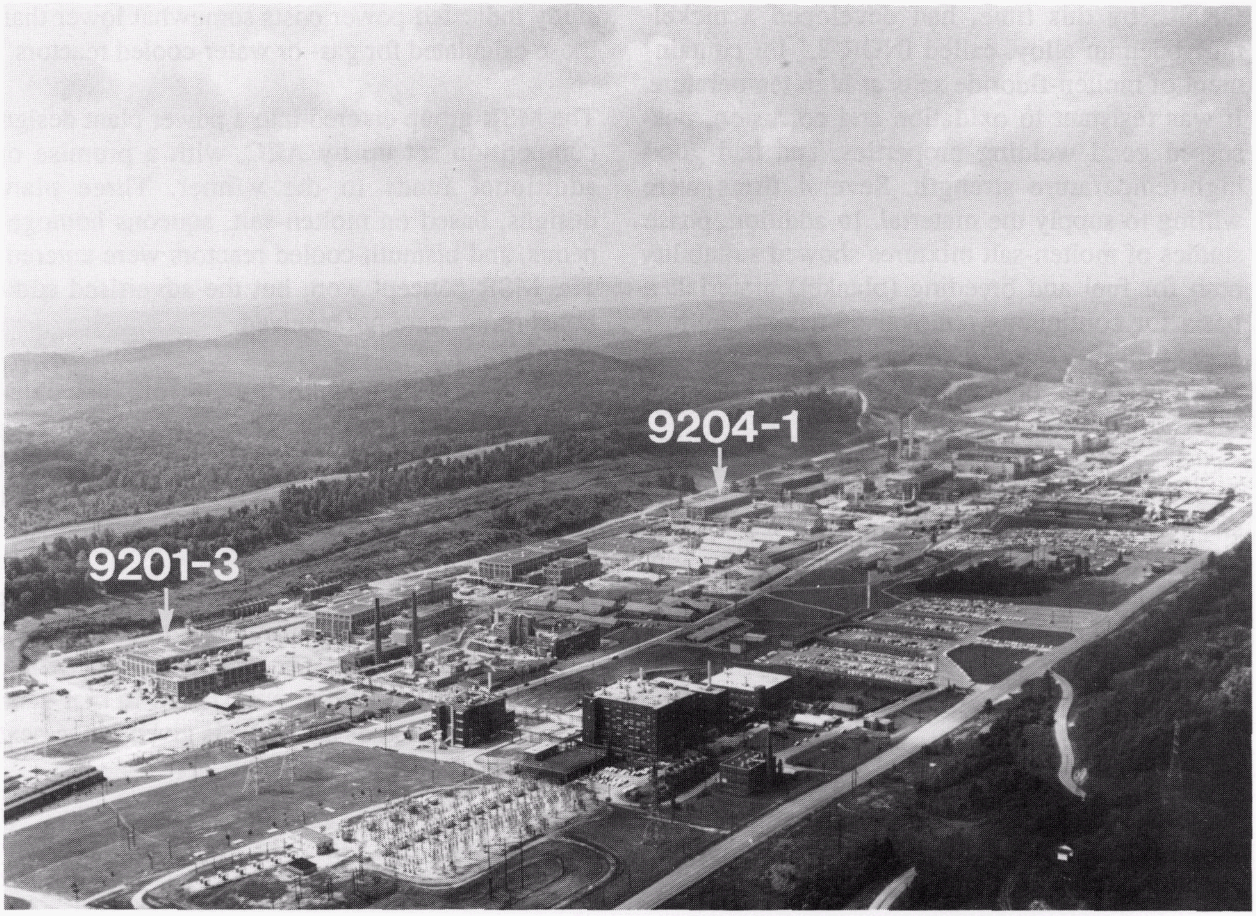
A conceptual design study of a power reactor showed a feasible arrangement of a reactor cell equipped for remote maintenance, suitable methods of handling off-gas, and arrangements for draining and otherwise handling the fuel. A cost study indicated power costs somewhat lower than those calculated for gas- or water-cooled reactors.

*This was actually a joint development by International Nickel Company and ORNL and is known commercially as Hasteloy N.

The MSR group entered into a power plant design competition set up by AEC, with a promise of additional funds to the winner. Three plant designs, based on molten-salt, aqueous homogeneous, and bismuth-cooled reactors were entered. The MSR concept won, but the advertised additional funds were not received.

Due to lack of funding, the initial work on a civilian MSR was significantly reduced before being rejuvenated at a later date. H. G. MacPherson was appointed Thermal Breeder Reactor Program manager under A. M. Weinberg in mid-1959.

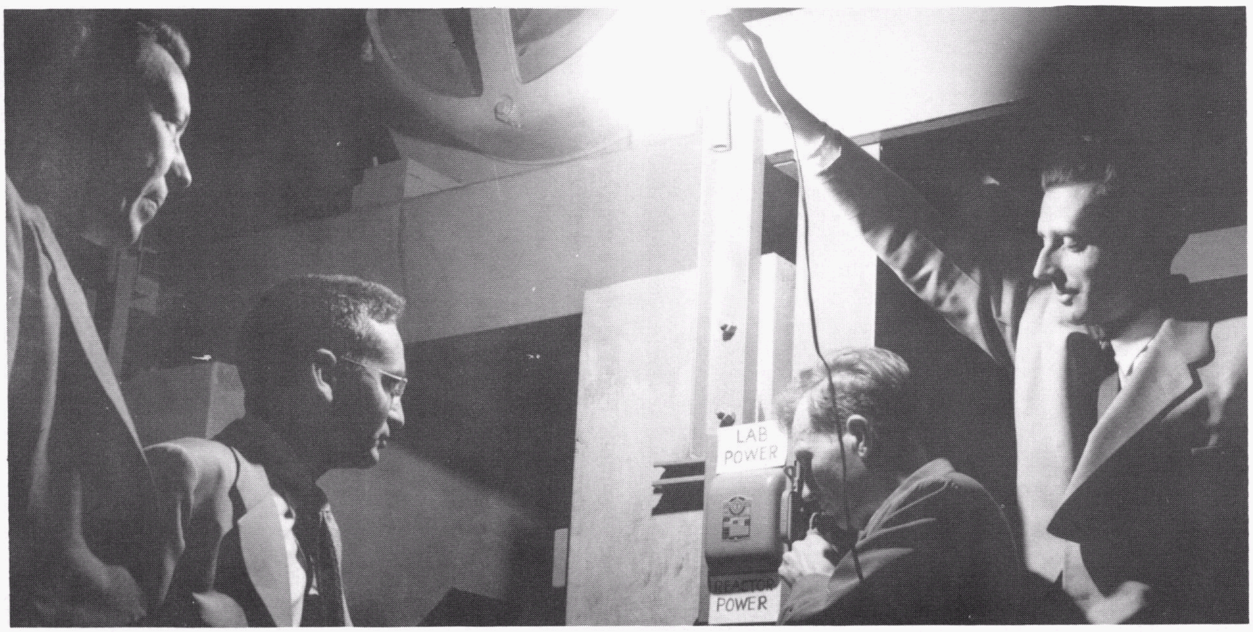
A. L. Boch was selected as Molten Salt Reactor Experiment (MSRE) project engineer in 1960, when increased financial support was forthcoming, and design of the MSRE was started in that year. Further discussion of this topic is given in the next section.




***In January 1947, the wartime Electromagnetic Plant, located at the Y-12 Site, was shut down after a brief, but intense, period of operation during which weapons-grade uranium was produced, allowing construction of a functional atomic bomb that would dramatically decrease U. S. losses during World War II. In 1951, the two newly formed ORNL reactor divisions, Aircraft Nuclear Propulsion and Reactor Experimental Engineering, moved from X-10 to Y-12 to occupy two of the large vacated process buildings. The Aircraft Nuclear Propulsion Division occupied Building 9201-3, referred to as $\alpha$-3, and the Reactor Experimental Engineering Division occupied Building 9204-1, which is referred to as $\beta$-1.***

*The Homogeneous Reactor Experiment (HRE) was a pilot-plant model liquid-fuel reactor built to investigate the chemical feasibility of maintaining a nuclear chain reaction at temperatures and power levels sufficiently high for production of electricity. Full-power operation was achieved on February 24, 1953; the reactor was dismantled in 1954.*

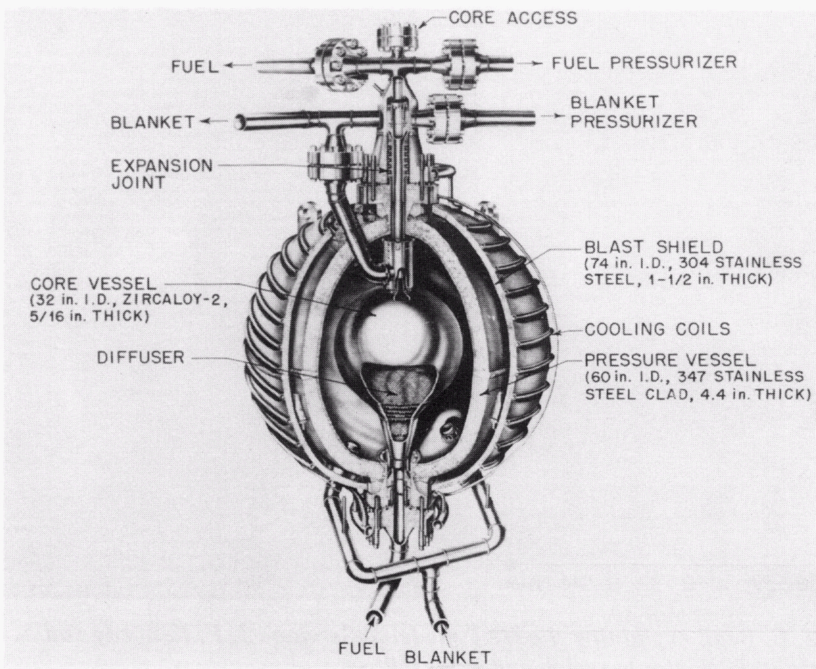
*Startup of the HRE, February 23–24, 1953. At the console: P. M. Wood; left of console: J. J. Hairston, J. W. Hill, R. L. Moore, S. I. Kaplan, J. A. Ransohoff, and S. E. Beall; right of console: J. A. Swartout, A. M. Weinberg, S. Visner, C. E. Winters, L. R. Quarles; back wall left to right: J. L. Redford, A. L. Johnson, R. W. Keller, and V. K. Pare.*




*The quartet of ORNL scientists and engineers primarily responsible for the HRE relax after the reactor successfully began producing 150 kW of electric power, enough to light the reactor building and feed a substantial amount back into the Laboratory's power system. Left to right are J. A. Swartout, A. M. Weinberg, S. E. Beall, and C. E. Winters.*

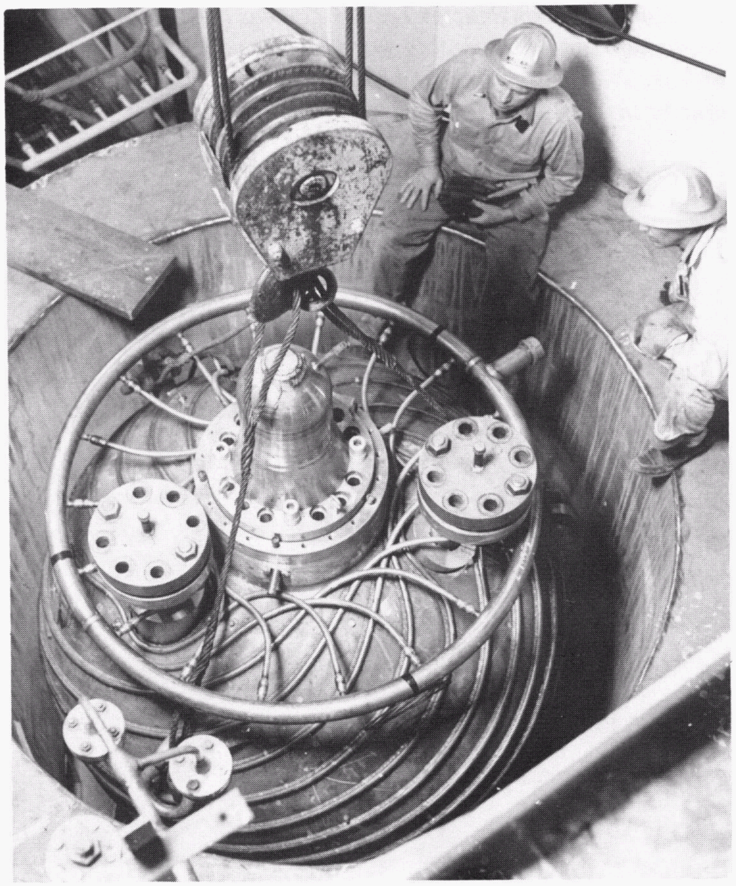
***The Homogeneous Reactor Test (HRT) was built to take the second step toward a full-scale power station. Fuel production through breeding was also planned. Reactor criticality was reached on December 27, 1957. Because of leaks in the core vessel, the reactor was operated intermittently. These leaks led to permanent shutdown of the reactor in April 1961.***

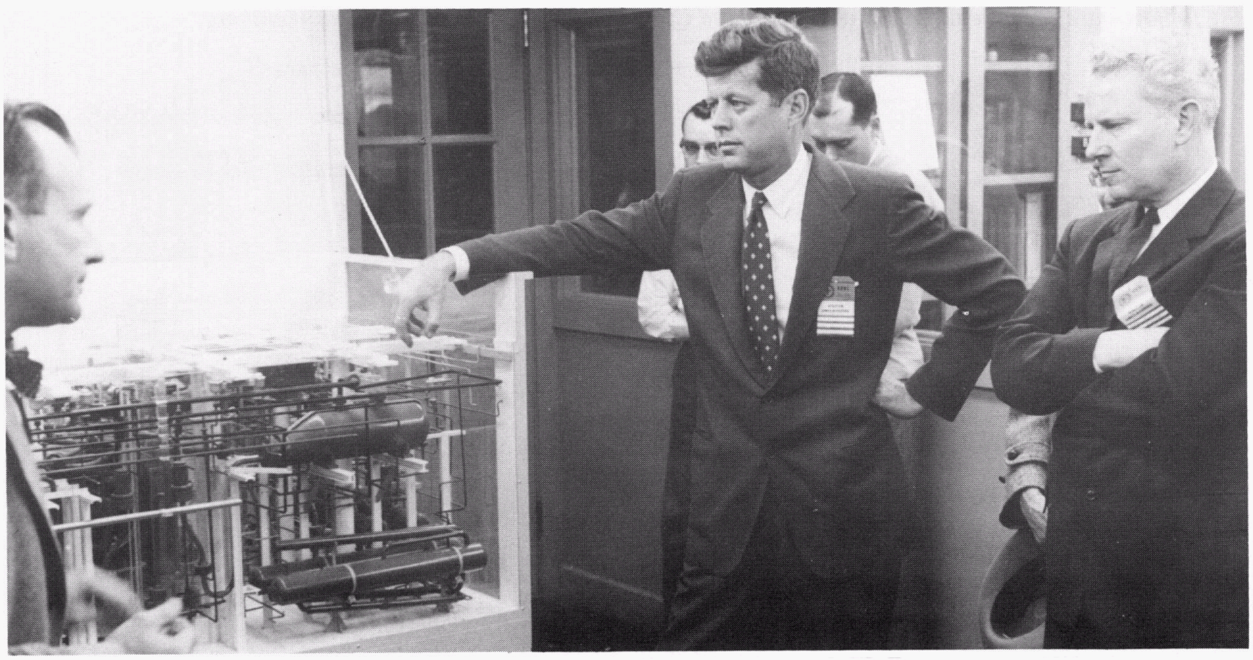
***Startup of the HRT on December 27, 1957. Sitting left to right, P. N. Haubenreich and R. B. Briggs; standing in the first row are S. E. Beall, A. M. Weinberg, and C. E. Winters; J. O. Kolb is behind Weinberg.***




***Schematic cut-away view of the blast shield, pressure vessel, and core tank of the HRT (or HRE No. 2). Also shown is the diffuser section. It was in the region of this diffuser section that leaks occurred and compromised the operability of the reactor.***

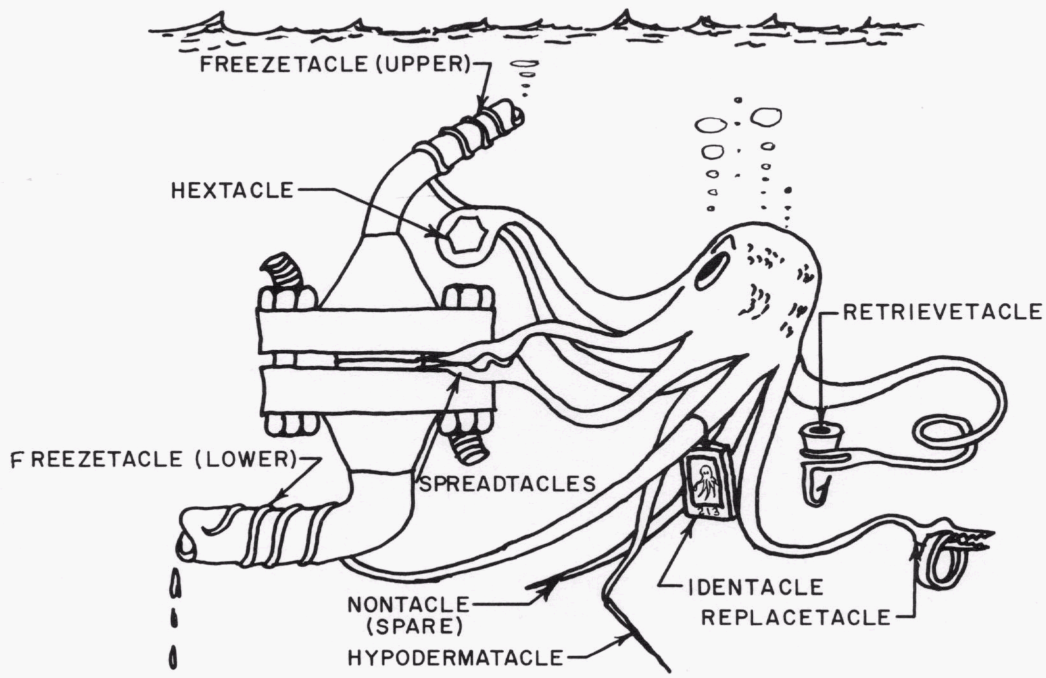
*HRT reactor pressure vessel assembly installed in the reactor chamber. This assembly included an outer blast shield, the pressure vessel, core tank, and other components.*





***S. E. Beall explaining the HRT model to Senators J. F. Kennedy and A. Gore, Sr., in 1958. Background left to right: P. N. Haubenreich and J. W. Hill.***

*Technicians at work on control and read-out equipment of the HRT. From right to left, J. Eves, H. Roller, and J. Wolfe. The Homogeneous Reactor Project had a relatively large pool of technicians from which individuals were dispersed to other activities throughout the Reactor Division in the waning period of this project.*

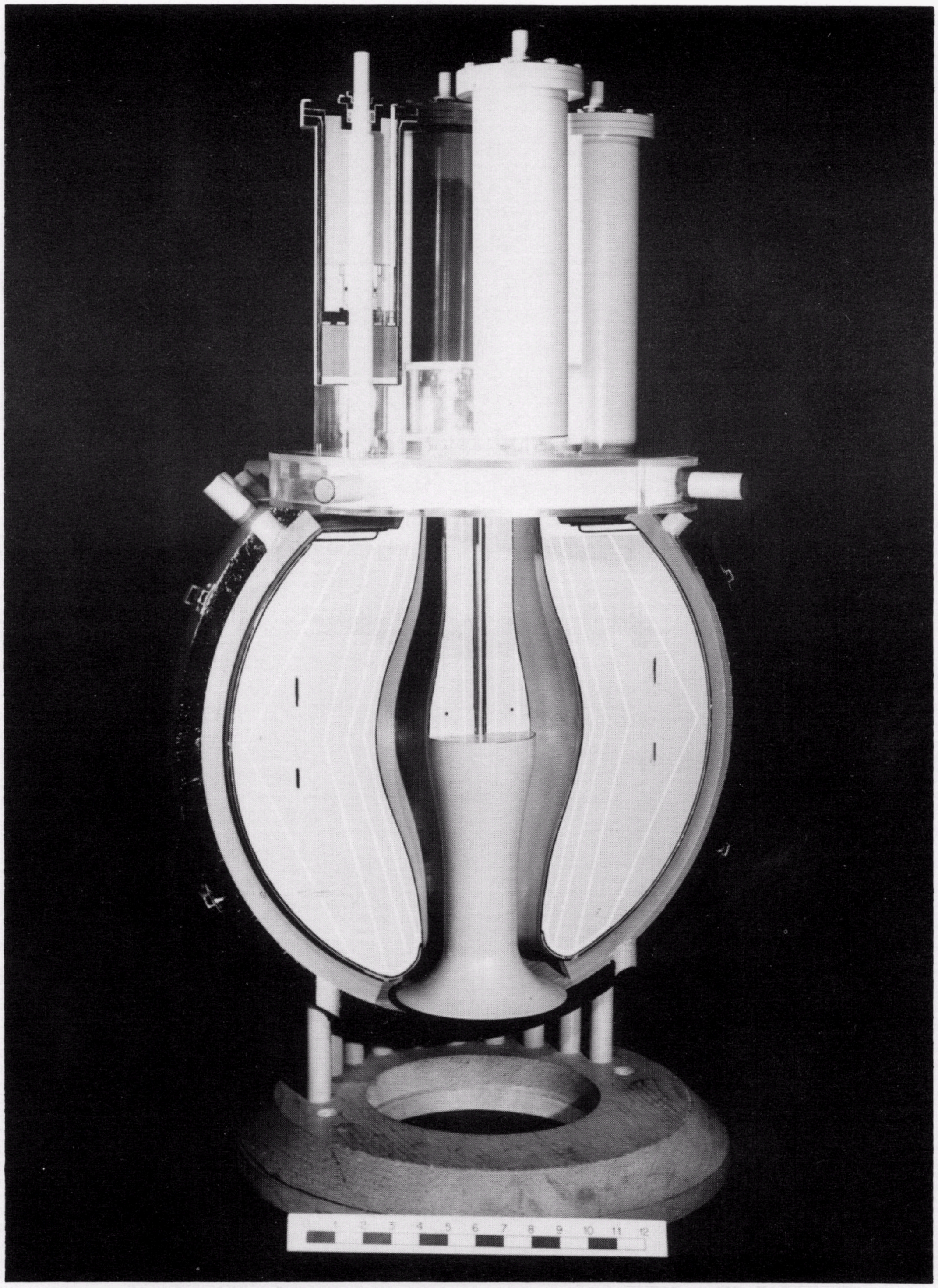



*Humorous cartoon of a multiple-purpose octo-tool for underwater maintenance of homogeneous reactors.*

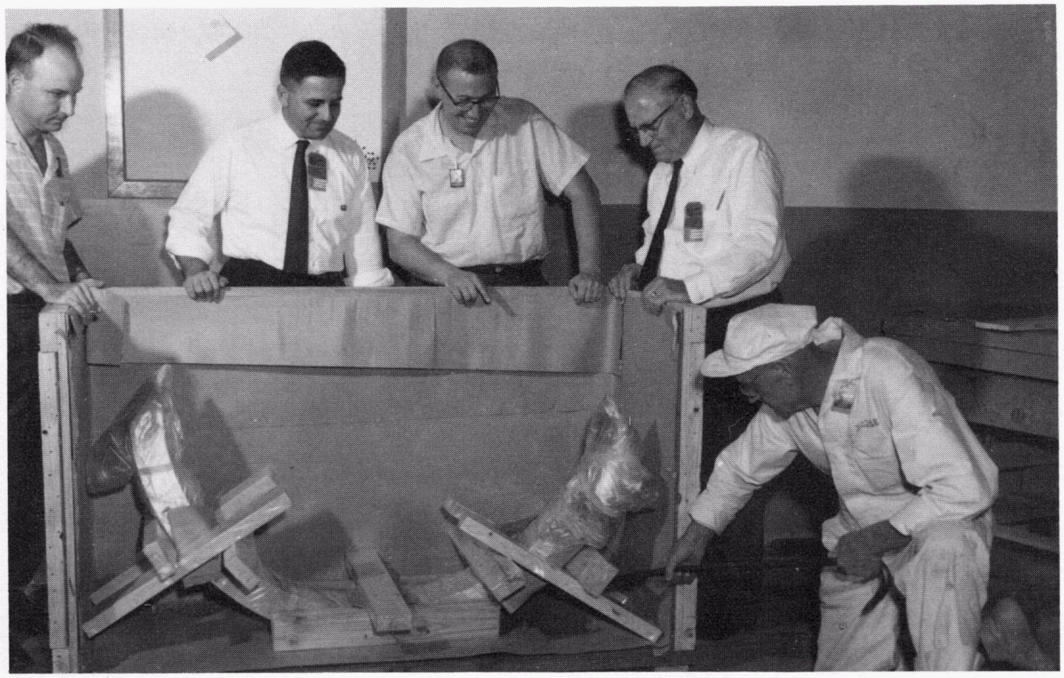
*The Aircraft Reactor Experiment (ARE) represented a first step in developing a liquid-fueled reactor for propulsion of aircraft. The ARE was to demonstrate the use of a liquid-fueled reactor at temperatures required for propulsion service. The reactor reached criticality on November 3, 1954. Specific operating objectives were to attain a fuel temperature of 1500°F, with a 350°F rise in temperature across the reactor and to operate for ~100 MW-h. All objectives were met, and ARE operation was brought to a close on November 12, 1954.*

*This picture was made as the reactor was being shut down. From left, E. R. Mann, Sylvan J. Cromer, Ed Bettis, USAF Col. Clyde Gasser, and J. L. Meem.*

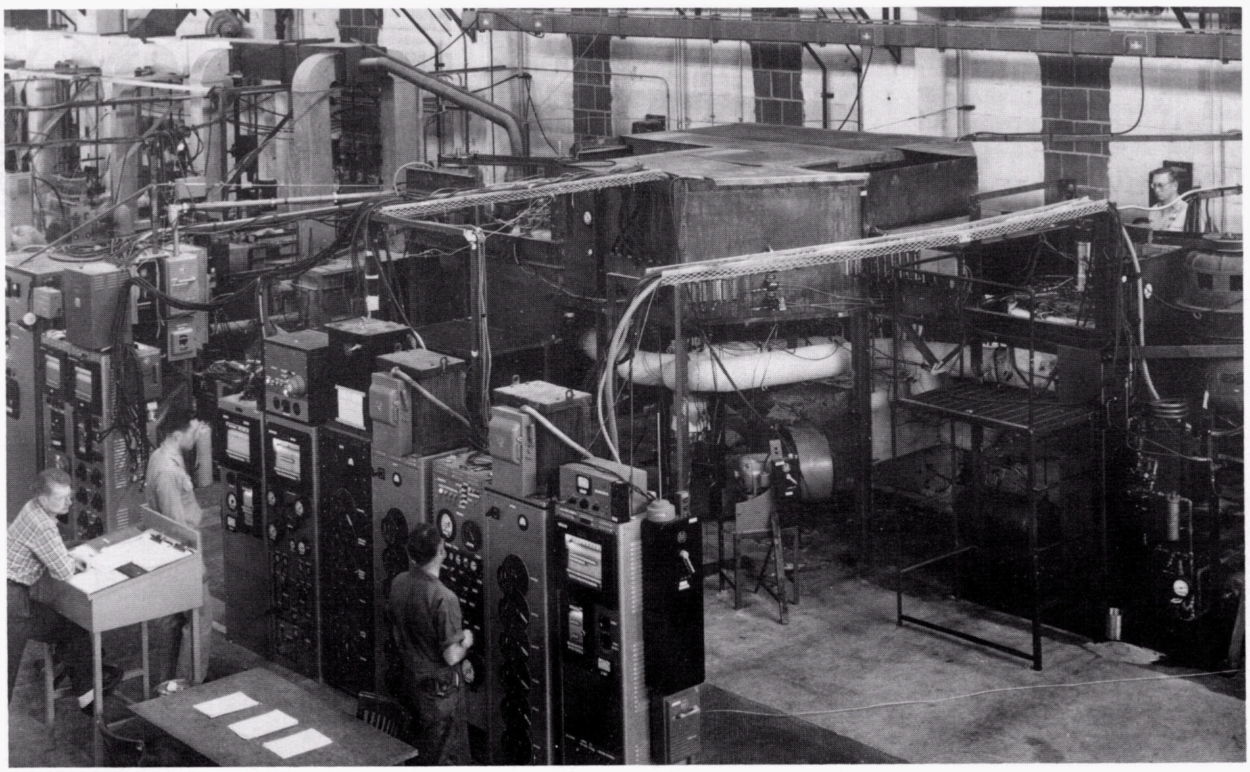
***The Aircraft Reactor Test (ART) was to provide the second step in liquid-fueled reactor development for propulsion of aircraft.***



***Sectioned model of the ART reactor showing the spherical vessel, beryllium reflector-moderator and island, fuel passages, and pumps (both sodium and fuel).***

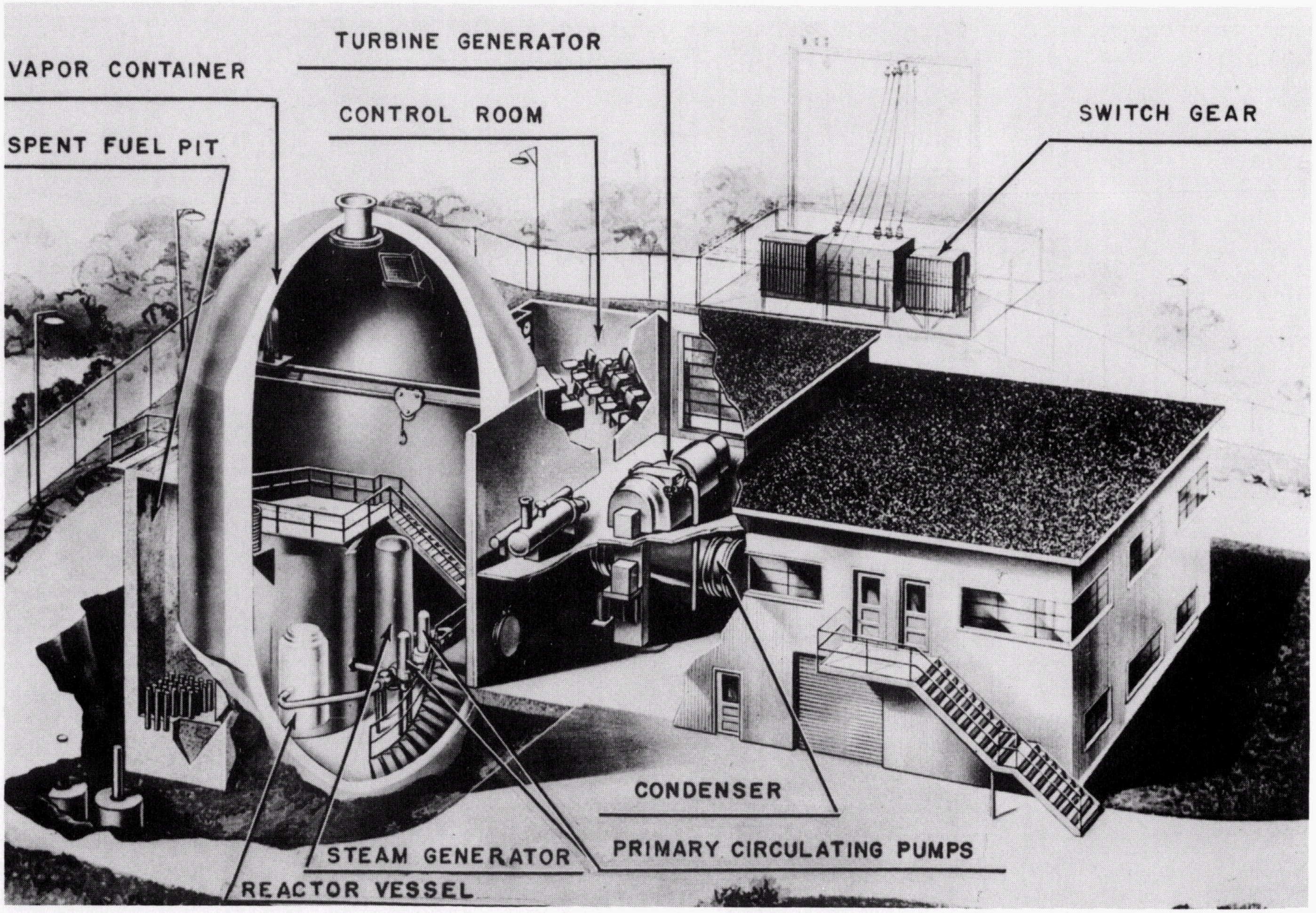


*Unpackaging a $1 million, fuel-to-liquid sodium-potassium (NaK), heat-exchanger bundle built by Black, Sivals, and Bryson (BSB) for the ART. Left to right: G. D. Whitman, BSB representative, M. Bender, BSB representative, and a Y-12 craftsman.*



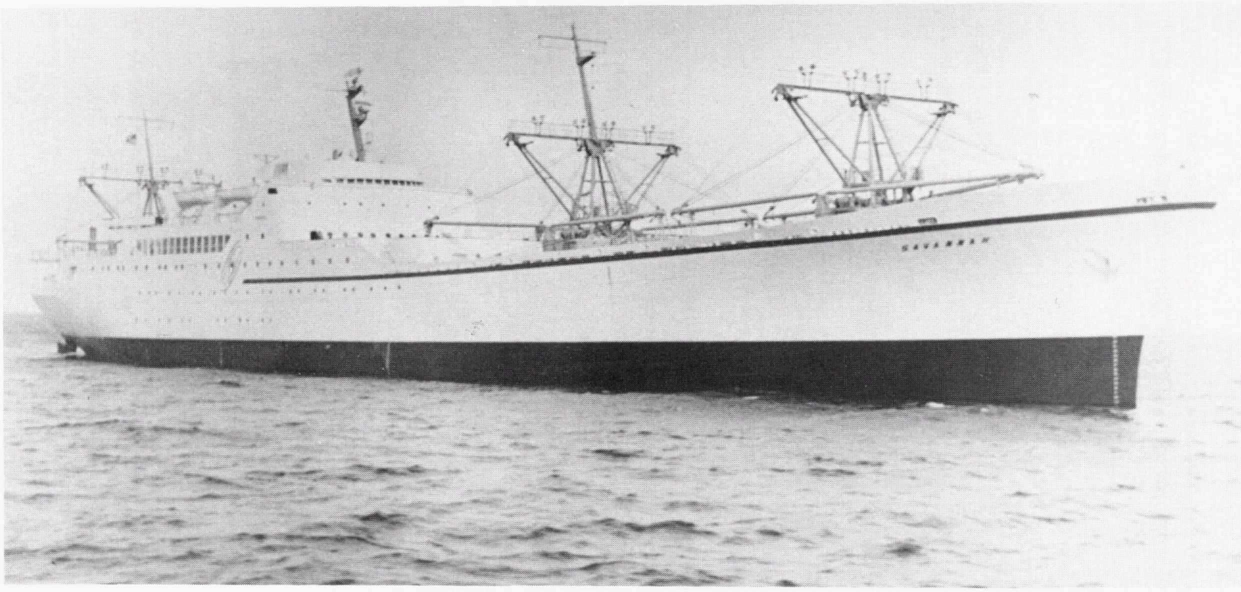
*First small heat exchanger test for the ART; from left to right, R. E. MacPherson, unidentified technician , J. R. Shugart, and R. Love.*

***The Army Package Power Reactor (APPR-1) was a prototype reactor designed to meet heat and power requirements at a remote military base. The conceptual design work for the APPR-1 (later designated SM-1) was done by ORNL.***

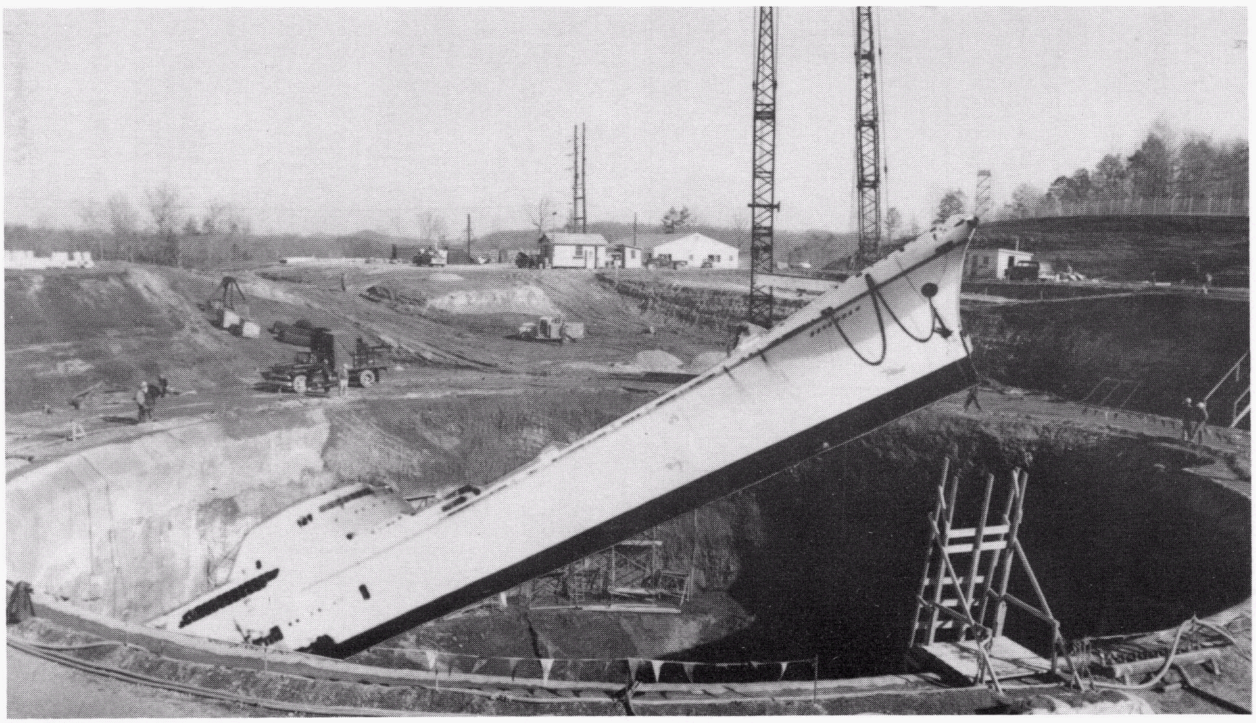



***Cut-away view of the SM-1 building showing the reactor system.***

*The N. S. Savannah was a nuclear-powered ship built to demonstrate peaceful uses of atomic energy in the world. It was a part of President Eisenhower's Atoms For Peace Program, as were the Geneva Conferences held in 1955 and 1958. The Savannah was launched by Mrs. Eisenhower on July 21, 1959, and was decommissioned in 1971. ORNL provided review and advisory services during the design, construction, and testing phases. In addition, ORNL aided in obtaining port clearances throughout the total span of operation.*



*The N. S. Savannah during initial sea trials in 1962.*



*Maximum credible accident showing N. S. Savannah lodged in excavation for EGCR containment building; presented to R. A. Charpie on his departure from ORNL in 1961.*


***A large ANP project group from ORNL and the U.S. Atomic Energy Commission (AEC) traveled to Wright Patterson Airfield in 1957 aboard this Air Force plane and, as a safety measure, the AEC's DC-3, "Delayed Neutron." Pictured, from left are two Air Force officers; Walt Jordon; an Air Force officer; Ralph Schultheiss; George Watson; Sylvan Cromer; Jim White; Bud Perry; Charley Barton; Dale Magnuson; Don Trauger; Bill Larkin; Mike Bender; Bob Megreblain; Ray Mann; Herb Hoffman; Randall Shields; Grady Whitman; unidentified; Al Taboda; Bill Ferguson; Wilfort Goode; Earl Breeding; Wes Savage; Air Force host; John Page, co-pilot of the AEC's DC-3; Bill Browning; Hap Wilson, pilot of the AEC's DC-3; Col. Jim Hill, Air Force officer assigned to the ANP Program in Oak Ridge and who later became deputy manager of the Oak Ridge Field Office; and Bob Affel.***

*Four key division participants in the Aircraft Nuclear Propulsion (ANP) Program are shown at a 1956 dinner sponsored by Pratt and Whitney Aircraft's CANEL (Connecticut Aircraft Nuclear Engine Laboratory) Division. From left are Bob MacPherson, Grady Whitman, Al Grindel, and Art Fraas.*

*Pratt and Whitney sponsored annual dinner dances for ANP Program participants in the late 1950s. This picture, made at Deane Hill Country Club in Knoxville, includes Bob MacPherson (facing camera at left) and wife, Claudia. To MacPherson's left are Al and Dot Smith. Further to MacPherson's left (with back to camera) is Bill Cottrell.*

*In 1957, a young Tom Kress validated the coolant flow distribution through one hemisphere of the beryllium reflector of the ANP reactor. The 57°F temperature of the test water helps explain the pained expression.*

*In the late 1950s and early 1960s, division personnel became increasingly involved in meetings and exchanges with foreign counterparts. In this 1959 photo, Hans Kronberger, Director, Research and Development, Engineering Group, Atomic Energy Research Establishment, Risley, England, meets with Reactor Projects Division personnel. Standing, left to right, are Bud Perry, Jay Foster, and Grady Whitman. Seated from left are Mike Bender, Bill Cottrell, Bill Greenstreet, Bob Charpie, Kronberger, and Garland Samuels.*

*This 1961 photo shows three visitors from France meeting with Reactor Division personnel. Left to right are Jacques Pelce, Engineer, Reactor Studies Department, Saclay, France; George Kidd; Raphael Meunier, Engineer, Reactor Studies Department, Saclay, France; Bill Cottrell; Paul Gelin, Engineer, Reactor Studies Department, Saclay, France; and Herb Hoffman.*

***A United Kingdom team discusses issues associated with the graphite core of gas-cooled reactors with Reactor and Metallurgy Division personnel in 1961. Seated, left to right, are Edwin Wood, Section Leader, Graphite Section, The English Electric Company, Ltd., Wetstone, England; Bill Greenstreet; Alan Littlejohn, Head, Materials Laboratories, The English Electric Company, Ltd, Whetstone, England; and Joel Witt. Standing are Jim Corum; Chuck Preskitt; Ray Kennedy, Metallurgy Division; and Sam Moore.***

***Division personnel regularly participated in various ORNL-sponsored sports tournaments, and sometimes were even winners. Here, Herb Hoffman and Jim Lane congratulate each other on winning the doubles tournament in bowling.***

# 4. 1961–1975 GLOBAL APPROACH TO NUCLEAR ENERGY (Reactor Division)

The period from 1961 to 1975 brought many changes in projects and outlook. Most significantly, the relatively large projects of the 1951 to 1961 era were replaced with projects or programs that became progressively smaller in size but larger in number. Although this period started with major attention given to reactor and reactor system design, development, and construction, it ended with more in-depth examination being given to reactor operation and safety. At the same time, the focus began to broaden to address energy resources in general as well as all aspects of nuclear power use. In addition, emphasis on management practices and matrix management was intensified as projects became smaller.

Safety and environmental concerns drew increased attention, abetted by utilities building nuclear plants to produce electricity. Hence, a general awareness of liabilities as well as benefits of nuclear energy and its products began to emerge.

Early in this period, the Reactor Division was at a low point because of reductions in activities and funding in 1960 and 1961. These circumstances forced a reduction in staff in 1961.

Continued efforts were made to expand existing projects and to obtain new ones. Thus, following the early low, the division grew during the 1960s, reaching a maximum of 288 people in 1969. However, at the end of the decade, support began to decline, and a second reduction in staff was necessary in 1971. Approximately 50 people were transferred, placed on loan, or terminated at that time.

R. A. Charpie left Oak Ridge National Laboratory (ORNL) to assume a position at Union Carbide Corporation headquarters in New York City in July 1961. H. G. MacPherson, who was then Associate Director of ORNL, became Acting Division Director. He continued in that capacity until 1963, when S. E. Beall was appointed Division Director.

The Design Section of the division was integrated into the multiplant Engineering organization in 1973. In 1974, a new division, the Energy Division, was formed, and some Reactor Division personnel and projects were shifted to the new division. S. E. Beall was appointed Director of this new division, and G. G. Fee became Director of the Reactor Division. Atomic Energy Commission (AEC) nuclear safety work, including that being done at ORNL, was transferred to a new organization, the U.S. Nuclear Regulatory Commission (NRC), which was formed in 1974.

With the shift in programs from the AEC to NRC, projects in the work for others category* underwent a dramatic increase. Other organizations that supported work during the 1961 to 1975 period were the Department of Housing and Urban Development, Department of Interior Office of Saline Water (OSW) and Office of Coal Research, and the Department of State Agency for International Development (AID).

During the latter part of the 1961 to 1975 period, emphasis on reactor system, hardware, and plant development declined. In the main, there was a pronounced shift to small, diffuse undertakings. Closer relationships between sponsors, project managers, and personnel increased ubiquitously. Thus, the influence of outside authority in addition to ORNL authority, although always present, became increasingly prominent as well as, at times, conflicting. Micromanagement became a well-known phenomenon. Nevertheless, important technological gains were made. Division organization charts for 1963 and 1973 are provided in Appendix B.

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*Work for sponsors other than AEC.

## 4.1 GAS-COOLED REACTOR PROGRAM*

As stated previously, there were two major parts to the Gas-Cooled Reactor Program (GCRP): the Experimental Gas-Cooled Reactor (EGCR) Project and the Advanced Reactor Development Project. Each is discussed below.

A number of changes occurred in GCRP leadership during the 1961 to 1975 period. When R. A. Charpie left ORNL, W. D. Manly, who was associate program director, became program director. In 1962, M. Bender was appointed head of the Design Section in the Reactor Division, and G. D. Whitman was selected to head the ORNL EGCR Project. Starting in about 1963, major program and project directors reported to an associate or assistant ORNL director, as in the case of the division director.

On April 1, 1964, W. D. Manly left ORNL to assume a position at Union Carbide Corporation headquarters in New York City. The GCRP was then divided into three parts: G. D. Whitman was named director of the EGCR Program, D. B. Trauger became director of the Advanced Gas-Cooled Reactor (AGR) Program, and M. Bender became responsible for Gas-Cooled Fast Reactor (GCFR) work. Soon thereafter, the GCFR effort was merged into the AGR program. This latter program included foreign exchanges other than the EGCR agreements with the United Kingdom.

In 1966, the AGR Program became the GCRP, with D. B. Trauger as director. In 1970, P. R. Kasten replaced Trauger as director of the GCRP.

### 4.1.1 EGCR Project/Program

Work was continued on the EGCR with the aid of several people from the Tennessee Valley Authority (TVA) who were to be on the plant operating staff. As work progressed, the need for an emergency core cooling system (ECCS), not included in the original design, became obvious from a safety standpoint. Responsibility for the detailed design of this system and procurement of associated equipment, as well as installation and testing, were assigned to Union Carbide Nuclear Company. The work was done by K-25 personnel under R. W. Ulm, with review responsibilities assigned to ORNL.

This system was to flood the reactor primary system with nitrogen for quickly reducing the temperature to a safe level. Design was initiated in 1962, and tests of the installed system were completed in the spring of 1964.

Areas of work being addressed at this time were advanced reactor design, physics, safety, graphite properties and behavior, fuel and associated materials, component and system development, stress and heat transfer analyses, and irradiation effects on materials. Altogether, about 580 reports, memorandums, and other publications were produced. The documents included in this total were issued by ORNL, TVA, Kaiser Engineers, and Allis Chalmers Manufacturing Company. Reactor Division participants having lead roles are listed in Table 4.1.

Because the EGCR was an experimental, prototypical, power reactor, safety considerations were very important and were addressed in a broader sense than was previously done. Existing philosophies dealing with hazards of water-cooled power reactors provided an inadequate basis for formulating a realistic hazards appraisal for EGCR-type reactors. In addition, precedents and criteria to serve as guidelines for adequate and safe design and operation were not available. As a consequence, the EGCR received a much more intensive and exhaustive review of hazards problems than had been typical of other experimental reactors.

In total, work on the EGCR project was technology-development intensive in essentially all areas addressed. This experience was extremely useful in projects that followed.

*The name was changed from project to program in 1961.

**Table 4.1. EGCR Project Personnel (Reactor Division) 1961 to 1966**

| Name | Responsibility |
|---|---|
| A. M. Perry<br>C. A. Preskitt<br>E. A. Nephew | Reactor Physics |
| G. Samuels<br>W. B. Cottrell<br>T. H. Row<br>J. O. Kolb<br>R. E. Helms<br>R. E. MacPherson | Performance Analyses |
| W. L. Greenstreet<br>F. J. Witt | Structural Analyses |
| H. W. Savage<br>R. E. MacPherson<br>R. E. Helms<br>F. H. Neill<br>E. R. Taylor<br>W. F. Ferguson<br>W. F. Boudreau<br>A. B. Meservey<br>J. W. Michel | Component Development and Testing |
| Members of Metallurgy Division | Materials Development |
| D. B. Trauger<br>H. C. McCurdy | Irradiation Testing of Components and Materials |

By January 1966, the EGCR was nearing completion; fuel loading and shakedown operations remained. Operating shifts were set up, and approval had been received for 20% power operation at a minimum coolant flow of 50%. Extension of this power level was contingent, among other things, on operational experience at the lower power levels and further review of the effectiveness of certain engineering safeguards incorporated into the system.

On January 7, 1966, AEC announced that it was terminating the EGCR Project. Factors cited as important to this decision were continuing design and engineering difficulties, with corresponding delays and rising costs, the diminishing potential of timely and significant contributions of the EGCR Project to commercial development of high-temperature gas-cooled reactor (HTGR) technology, and competing demands for limited funds.

Several factors can be identified as contributors to this action. These were lack of competitiveness with light-water reactors (LWRs), increased emphasis on LWR safety, liquid metal fast-breeder reactor (LMFBR) work, and the Peach Bottom reactor, with its advanced core design, which was nearing start-up.

### 4.1.2 Advanced Reactor Development

Work in this area began with a scoping and study phase, followed by work addressed to fixed reactor concepts. Both phases are addressed here.

#### 4.1.2.1 Scoping and Study Phase

The early GCRP embraced studies of AGRs, including pebble-bed reactors (PBRs). In 1959, W. B. Cottrell and others published the results of a study on a high-temperature reactor system. This study was based on the British Calder Hall and large Central Electricity Authority plant layouts, as was the GCR-2 study. The reactor, in this case, incorporated ceramic fuel elements, that is, graphite-contained uranium oxide elements. This study implied that fission-product release to the coolant gas stream would not be so great as to make maintenance problems unmanageable for a helium-cooled, graphite-moderated reactor system. However, other changes were needed to reduce capital costs relative to the GCR-2.

Therefore, additional studies published in 1962 were conducted by A. P. Fraas and others for various reactor core and steam generator arrangements to reduce costs. Ceramic fuel elements were assumed together with either a graphite or a beryllium oxide moderator. Both helium and carbon dioxide, as used by the British, were used as coolants. Results from these studies implied that it might be possible to build a large gas-cooled, all-ceramic-fueled and moderated plant for less than a coal-fired plant. These studies were intended for guidance in future developments and were followed by others as described below.

The PBR concept is an old one; it was used in the Daniels Pile. As the name implies, the fuel region is in the form of a pebble bed, being made up of packed fuel elements.

Advantages of this type reactor are simplicity of the fuel element, ease of fuel handling, on-line fuel processing by element replacement, suitability for high-temperature operation, and good neutron economy. These desirable attributes engendered a need for exploration.

Sanderson and Porter, with AEC support, began to investigate the PBR design and development in about 1957. Most of this effort was devoted to the study of fuel handling problems and to the development of a fuel for the reactor, with aid from the Battelle Memorial Institute, which was working on coated particles.

A German combine of the Brown-Boveri Company and Krupp began, in 1956, to actively develop a 15-MW(e) PBR and to plan for its construction. The result was the Arbeitgemeinschaft Versuchs-Reaktor (AVR).

In the fall of 1960, ORNL was requested by AEC to perform design studies of both an experimental reactor and a large power reactor based on the pebble-bed concept. These studies were completed by A. P. Fraas and coworkers in 1961.

Further studies of a PBR experiment (PBRE) were initiated at ORNL under the direction of M. W. Rosenthal. The outgrowth of these studies was a report, published in 1962, which contained a preliminary description of the design for the reactor experiment. The objectives were (1) to investigate key features of the pebble-bed concept, including on-stream fuel handling, movement of fuel through the bed, and performance of the core; (2) to obtain operation and maintenance experience with a system contaminated with fission-product activity; and (3) to investigate the behavior of graphite fuel elements. A fourth objective, study of the behavior of core materials at conditions occurring with exit gas temperatures in the range 2000 to 2500°F, was tentatively included.

The preliminary design to achieve these objectives was that of a 50-MW(t) reactor. The core of the PBRE was a 2.5-ft-diameter, 4-ft-tall cylinder containing ~12,000 spherical graphite fuel elements with 1.5-in. diameters. Fuel spheres were to be added to and removed from the core by gravity flow; these operations were to be performed with the reactor at power. The helium coolant was to enter the bottom of the core at 500-psi pressure and 550°F and to emerge from the top at 1250°F.

A research and development (R&D) program needed to ensure satisfactory performance of the PBRE was defined. However, many of the studies discussed were already being conducted as continuing programs for the general development of GCR technology. Therefore, from the areas identified, the following were pursued: graphite (both fuel and moderator) reactions with impurities; fuel development; core cooling problems (flow through and heat transfer within the bed); and graphite irradiation effects, with graphite irradiation-induced dimensional change work being done at Hanford. Fuel development embraced experimental work on coated-particle fuel.

A team from the German AVR project subsequently visited ORNL to exchange information on the PBR concept. The fuel elements for the AVR, at that time, were to be made up of uranium carbide fuel in a graphite circular cylinder on the order of 2 in. in diameter and 3 to 4 in. long. However, ORNL experiments conducted in the Materials Testing Reactor (MTR) on similar elements showed that such elements could swell and distort badly from relatively short irradiation exposure. Consequently, the need for a change was evident. An outcome was an AVR Memorandum of Understanding, negotiated between the AEC and the German Ministry, that allowed ORNL to participate in the development of fuel elements for the AVR.

Because General Atomic Company was developing the HTGR with support from both the AEC and utilities and it was eventually determined that the PBRE was not cost-effective, the PBRE project was canceled in 1963. However, the coated-fuel development and fission-product behavior studies were continued. In effect, work on the PBRE evolved into a support project for AVR fuel development.

The GCRP became primarily a fuel development program in 1966, with involvement by the Metals and Ceramics (formerly Metallurgy), Reactor Chemistry, and Reactor Divisions. The role of the Reactor Division, in this case, was that of designing, constructing, and conducting irradiation experiments. Early irradiation-experiment activities in the Aircraft Reactor Engineering/Reactor Projects/Reactor Division were under the direction of D. B. Trauger with assistance from J. A. Conlin and others. H. C. McCurdy replaced Trauger as director of these activities in 1964, when Trauger became director of the AGR Program.

The first AVR fuel elements used were hollow spherical balls made of graphite that encapsulated uranium carbide fuel particles. These particles had two-layer (duplex) coatings of pyrolytic graphite for fission-gas retention. The fuel elements were manufactured by Union Carbide Corporation. Fuel elements with fuel particles that had advanced coatings to contain both fission-gas and metal fission products were used during the last years of the AVR's operating life.

The reactor, located at the Institute for Reactor Development in Jülich, Germany, reached criticality in 1966 and operated successfully over its lifetime. Although it was designed for a 15-MW(e) output, the output was actually 13.2 MW(e) due to heat exchanger limitations. The reactor outlet gas temperature was about 1740°F during the last years of operation. Operation was terminated in 1988 due to lack of a perceived mission.

A cooperative effort was initiated in 1961 with the European Organization of Economic Cooperation and Development Dragon Project for high-temperature reactor development. The focus was the Dragon reactor, a helium-cooled, graphite-moderated unit with a power output of 20 MW(t), which was built in England. A number of countries were involved, including the United States, the United Kingdom, Germany, the Netherlands, Canada, Australia, and, to a lesser extent, France. Extensive fuel element development work was done by ORNL, involving the Solid State Physics and Reactor divisions, which conducted irradiation experiments, and the Metallurgy Division, which addressed materials development.

The Dragon Project was very successful, with excellent cooperation between participants. The reactor first reached criticality in 1964, and cooperation between participants continued until the reactor was shut down in 1976. Cooperation between the United States, the United Kingdom, and Germany in the fuel development area was especially noteworthy.

Conceptual design and evaluation activities were continued in the Reactor Division. In 1963, the results of a study directed by M. Bender and W. R. Gall (with participants from ORNL, TVA, Combustion Engineering Inc., and Westinghouse Electric Corp.) was published. This study projected that the cost of electrical energy produced from a large [2000-MW(t)], commercial, EGCR-type reactor would be in the range then reported for fossil-fueled power stations in the continental United States.

A companion study was conducted to examine the influence of replacing clad fuel with ceramic fuel, a change that yields improved neutron economy and higher allowable reactor outlet temperatures. The higher gas outlet temperature translates into increased latitude in the heat-to-power conversion cycle selection. From this, it was projected that a significant 10% increase in overall plant efficiency could be realized.

Another study, completed in 1964, addressed gas-cooled fast reactor* (GCFR), or breeder reactor, concepts for commercial plants. The objective was to examine the feasibility and performance of reactors of this type. This study indicated that reactors having clad oxide fuel, with helium, carbon dioxide, or sulfur dioxide as coolant and working pressures of 1000 to 1500 psi, have a slightly higher potential for breeding than sodium-cooled, fast-breeder reactors. The latter were being pursued by AEC at that time.

#### 4.1.2.2 Fixed Reactor Concept Phase

Following termination of the EGCR project, reactor studies under A. M. Perry were reoriented to allow ORNL participation in the Peach Bottom HTGR Zero Power Commissioning Program. ORNL also participated in the surveillance of circulating radioactive material in the reactor coolant circuit and worked closely with General Atomic in predicting nuclear properties of the reactor.

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*Reactors are labeled thermal or fast, depending upon the energy (kinetic energy or speed) of neutrons causing most of the fissions.

Early in 1966, studies of large HTGRs were expanded to include evaluations of designs and of fuel cycle performance; these studies were done primarily by Reactor Division personnel. Work was continued on fueled graphite development and associated fission product behavior (Metals and Ceramics) as well as PBR-type spherical fuel elements. Investigations of moderator structures were continued, and prestressed concrete reactor vessel (PCRV) development planning work was initiated in 1965 under M. Bender, with the aid of D. B. Trauger and W. B. Cottrell. J. M. Corum played a lead role in defining the program elements, and a description of the proposed program was submitted to AEC in December 1965. AEC approval was obtained, and G. D. Whitman was given management responsibility in early 1966. Work on the PCRV Program will be discussed in Sect. 4.12.

Throughout the period from 1966 to 1975, fuel element development was a major item in the GCRP. It was primarily concerned with understanding, evaluating, and improving fuel performance and with studying the behavior of certain fission products, including their effects on fuel performance. This activity involved a significant amount of irradiation work by Reactor Division personnel and was largely addressed to the needs of advanced reactors, with the fuels corresponding to those for the Fort Saint Vrain reactor and to commercial HTGR designs, as well as to fuels for GCFRs. The reactors addressed were of General Atomic Company design. The first employed ceramic fuels, while the latter used metal-clad fuels. GCFR fuel irradiation studies emphasized fuel-cladding interactions among other aspects.

During this time, improved coatings for fuel particles were developed. In particular, a coating system, called the Triso coating, was developed to retain both fission gases and metals, including strontium and cesium.

Fission products, such as cesium and strontium, in the helium circuits of HTGRs pose potential maintenance and hazard problems because they plate out on metallic surfaces and on carbon dust. Cesium is especially troublesome because it has a relatively long half-life* and most of the fission product isomers emit energetic gamma rays. Therefore, fission product transport and deposition studies were conducted during the period after 1966 in the Reactor Chemistry and Reactor Divisions, with experimental studies being carried out by both divisions. R. E. MacPherson, F. H. Neill, D. L. Gray, T. S. Kress, and others were engaged in this work until mid-1968.

Reactor Division personnel, including F. H. Neill, R. E. Helms, and T. S. Kress, participated in studies of steam-graphite reactions until mid-1968 by conducting experiments and analyzing results, while Reactor Chemistry Division personnel did mathematical modeling. Such studies are particularly important because leakage of steam into the coolant stream of an HTGR can, for example, result in partial loss of the moderator graphite structure and in the generation of potentially explosive quantities of hydrogen and carbon monoxide.

In 1967 and 1968, additional component development work beyond that on PCRVs was initiated. Rotating seals for gas circulators were examined by F. H. Neill, E. R. Taylor, and coworkers; properties of nickel alloy weldments for HTGR steam generators were examined by Metals and Ceramics Division personnel; and steam generator designs were examined under A. P. Fraas.

Overall, various stress analysis, heat transfer, and other studies were done in support of the GCRP. Reactor analysis and assessment studies persisted throughout most of the period to 1975. Those who participated in this work included A. M. Perry, J. M. Corum, L. L. Bennett, H. W. Hoffman, R. S. Holcomb, J. D. Jenkins, G. J. Kidd, Jr., and G. T. Yahr.

Recognizing the need for a test facility to study the design steady-state and transient behaviors of

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*Half-life is defined as the time required for the radioactivity to decay to half its initial value.

fuel elements for the General Atomic GCFR, the responsibility for a test loop was assigned to ORNL. At ORNL, this test loop was known as the Core Flow Test Facility and was under the direction of U. Gat. During 1973, facility requirements were defined, and work on conceptual design was initiated. This closed-circuit, out-of-reactor loop was to circulate helium at temperatures and pressures anticipated in GCRs and at flow rates sufficiently large to perform engineering-scale tests. Facility construction and testing activities were under the direction of R. E. MacPherson A. G. Grindell, and H. C. Young

Although separate from the GCRP, other HTGR work was conducted at ORNL, including work under the Nuclear Safety Program and fuel recycle studies under the Thorium Utilization Program. The Thorium Utilization Program became part of the GCRP in 1970 when P. R. Kasten was named director. At that time, the HTGR portion of the GCRP consisted of coated-particle-based fuel development, materials studies, and HTGR fuel recycle development involving fuel reprocessing and refabrication. During the 1970s, the reactor technology program expanded into the HTGR safety area; in 1975, the safety work included both development and licensing-support activities. (HTGR safety work and the HTGR Safety Program Office are described in Sect. 4.6.)

The Thorium Utilization Program was initiated in 1962 under J. A. Lane and had, as its primary objective, the goal of developing low-cost uranium-233 recycle processes. A second objective involved studies of the feasibility and economics of thorium reactors.

The program evolved into the HTGR Fuel Recycle Development Program, covering reprocessing and refabrication. In general, fuel cycles for two different fuel types, that is HTGR and GCFR fuels, were considered. In the first case, the fuel was coated microspheres incorporated in a graphite matrix; in the second case, the fuel was oxide clad in metal tubes. This program was carried out almost exclusively by Chemical Technology Division and Metals and Ceramics Division personnel.

## 4.2 MOLTEN SALT REACTOR

Work on the Molten Salt Reactor Experiment (MSRE), which was begun in 1960 with A. L. Boch as project engineer, was continued. E. S. Bettis was in charge of design activities; he was aided by W. B. McDonald. In 1961, the MSRE was included under the Fluid Fuels Reactor Program (R. B. Briggs, Director) for the short time (less than a year) that this program existed. R. B. Briggs was made Director of the MSR activities when A. L. Boch became High-Flux Isotope Reactor (HFIR) Project Director in December 1961.

A major goal of the MSR work was to continue the quest for a breeder reactor. During the period 1957 to 1960, investigations were carried out at ORNL on fuel-salt chemistry, metallurgy of containment materials, design of salt-circulating pumps, and on remote maintenance techniques. Studies done in 1959 (by H. G. MacPherson, L. G. Alexander, and others) in combination with these investigations, led to a proposal to AEC to investigate remaining areas of uncertainty that could be resolved only by actually building and operating an MSR. In April 1961, ORNL received a directive from AEC to design, construct, and operate the MSRE.

A primary purpose of the MSRE was to investigate the practicality of the molten salt concept for central power station applications. In this role, the MSRE was envisioned as a straightforward installation, uncomplicated by the inclusion of experimental apparatus that might jeopardize reliable, long-term operation. It was also necessary that the MSRE be of large enough capacity for the experimental findings to be meaningfully extrapolated to full-scale plants. A reactor with 10-MW(t) output was selected to satisfy this criterion.

Conversion of the heat output to electricity was not considered necessary to demonstrate the concept; thus, the existing blowers and stack at the ART building were used to dissipate the heat to the atmosphere. Containment requirements dictated a double barrier between the highly radioactive fuel salt and the environment; hence, a secondary salt very similar to the fuel salt in composition and physical properties was chosen to transport the heat from the fuel salt to a radiator for transferring the heat to the atmosphere. All of this equipment was constructed of Hastelloy N.

The reactor vessel was a 5-ft-diameter, 8-ft-high tank that contained a 55-in.-diameter, 64-in.-high graphite-moderator core structure. This core structure was an assembly of vertical graphite bars 2 in. square by 67 in. long, mounted in a close-packed array. Salt entered the vessel at 1175°F and ~20-psi pressure; it exited at 1225°F and ~7 psi. The fuel pump provided a flow rate through the core of 1200 gal/min.

Fabrication of equipment, done by Y-12, began in 1962, and the reactor became critical on June 1, 1965. Success of the MSRE was considered crucial if ORNL was to convince AEC of the feasibility of molten salt breeders, which have a graphite-core structure, a uranium-bearing fuel salt, and a thorium-bearing blanket salt. These reactors would have a low breeding ratio, but an advantage was that of requiring less raw uranium for fuel.

Operation began at low power in January 1966, and sustained power operation was begun in December 1966. One run continued for 6 months, until the reactor was stopped on schedule in March 1968.

Completion of this 6-month run successfully ended the first phase of the MSRE operation. The objectives were to show, on a small scale, the attractive features and technical feasibility of the system for commercial power reactors. In addition to achieving this objective, the MSRE had shown that molten fluoride reactors can be operated at 1200°F without corrosion attack on either the metal or graphite parts of the system; the fuel is stable; the reactor equipment can operate satisfactorily at these conditions; undesirable fission gas can be removed rapidly from molten salts; and, when necessary, the radioactive equipment can be repaired or replaced.

The second phase of the MSRE operation began in August 1968 when a chemical process was used to remove the original uranium charge from the fuel salt. After the fuel was reprocessed, a charge of uranium-233 was added to the original carrier salt. On October 8, 1968, the MSRE became the world's first reactor to operate on uranium-233; it was brought to power by AEC Chairman G. T. Seaborg. In September 1969, small amounts of plutonium were added to the fuel to gain some experience with this material in an MSR. The MSRE was shut down permanently December 12, 1969, allowing funds supporting its operation to be used elsewhere in the R&D program.

In 1967, processes were developed to allow fuels containing thorium to be handled. This meant that the fertile and fissile materials could be constituents of the same salt. Hence, a one-fluid breeder, by making use of these processes, can have fuel utilization characteristics approaching those of two-fluid (separate fuel and blanket region) concepts. Because the graphite served only as a moderator, a one-fluid breeder power reactor would be close to a scale-up of the MSRE.

A conceptual design of a one-fluid, 1000-MW(e) breeder reactor was completed, but there was some question whether the project would be funded because of pressures of financing the war in Vietnam. The MSR Program was halted in 1973, but it was reinstated in January 1974.

The post-January 1974 program addressed conceptual design studies and work on materials, the chemistry of fuel and coolant salts, fission-product behavior, processing methods, and development of systems and components. Two important single achievements were as follows. The first was the development and demonstration of an alloy that has adequate resistance to tellurium-induced shallow cracking, as observed in Hastelloy N used in the MSRE. The second was the development of an adequate basis for safe management of tritium (a radioactive, hydrogen isotope) produced in the reactor. The program was terminated in 1976 because it was viewed as a source for diverting attention and resources from the LMFBR, which had been given first priority by AEC.

R. B. Briggs was Director of the MSR Program until M. W. Rosenthal was assigned to that role in 1966. P. R. Kasten became Deputy Director in that year. From 1966 through part of 1970 M. W. Rosenthal was Director, and R. B. Briggs and P. R. Kasten were Associate Directors; P. N. Haubenreich replaced P. R. Kasten as Associate Director in 1970. M. R. Rosenthal, R. B. Briggs, and P. N. Haubenreich remained in their positions until the program was halted in 1973.

Others who participated in the program included E. S. Bettis, R. B. Korsmeyer, W. B. MacDonald, and R. C. Robertson. In total, a number of people in the Reactor Division made significant contributions. Reactor analysis work was done under A. M. Perry by T. W. Kerlin, B. E. Prince, H. F. Bauman, W. R. Cobb, J. R. Engel, G. R. Ragan, O. L. Smith, and others; pump development work was conducted by A. G. Grindell and others; and component development was done by R. B. Gallaher, R. E. Helms, W. R. Huntley, A. N. Smith, P. G. Smith, H. C. Young, and others under H. W. Savage and R. E. MacPherson. Under I. Spiewak, D. Scott, R. Blumberg, E. C. Hise, P. P. Holz, R. J. Kedl, T. S. Kress, J. C. Moyers, R. P. Wichner, and others conducted development work. Designers under M. I. Lundin who contributed to the program included E. S. Breeding, C. W. Collins, W. K. Furlong, H. A. McLain, C. K. McGlothlan, J. R. McWherter, F. C. Zapp, and others. Reactor operations were the province of S. E. Beall followed by P. N. Haubenreich; the latter was aided by C. H. Gabbard, R. H. Guyman, J. R. Engel, M. Richardson, B. H. Webster, and associates.

L. E. McNeese was program director following reactivation of the program in 1974. Work carried out in the Reactor Division was under the direction of J. R. Engel. Specific areas addressed were design and system analyses (G. T. Mays, H. T. Kerr, E. J. Allen, J. M. Corum, G. T. Yahr, and others); system and component development (R. H. Chapman, J. C. Crowley, W. R. Huntley, A. N. Smith, M. D. Silverman, and others); and safety studies (E. S. Bettis and others).

## 4.3 HFIR

The HFIR was designed primarily as a part of an overall program to produce transuranium elements* for use in the heavy element research program of the United States. This reactor, with the world's highest thermal neutron flux, produces californium-252 and other isotopes by bombarding with neutrons such target materials as americium, curium, and plutonium. In addition to isotope production, this facility has served as an important base for neutron-scattering work needed in physics, chemistry, and other research. It also serves as a test reactor for studying irradiation effects on materials.

Associated with the HFIR is an elaborate processing plant (the Transuranium Processing Facility). This facility is used to chemically extract the newly produced, intensely radioactive californium and other heavy elements, such as americium, einsteinium, and fermium.

The HFIR is a beryllium-reflected, light-water-cooled and -moderated, aluminum-clad fuel plate reactor that utilizes highly enriched uranium-235 fuel. The design power level is 100 MW(t).

The reactor core consists of a series of concentric annular regions, each ~2 ft high. A 5-in.-diameter cylindrical hole exists at the center of the core. The target containing transuranium isotopes (plutonium-242 for californium-252 production) to

*Transuranium elements are chemical elements heavier than uranium, the heaviest element that occurs naturally on earth.

be bombarded is positioned on the reactor vertical axis within this hole. The fuel region is composed of two concentric fuel elements. The inner one, which contains 171 curved fuel plates, has an inside diameter of 5 in. and an outside diameter of 10.5 in. The outer fuel element contains 369 curved fuel plates and has inner and outer diameters of 11 and 17.134 in., respectively.

The HFIR is a so-called flux trap reactor. This is a reactor in which the core consists of an annular region of fuel surrounding an unfueled moderator region or "island." Such a configuration permits fast neutrons leaking from the fuel to be moderated in the island and thus produces a region of very high thermal neutron flux at the center of the island.

The fuel plates are 0.05 in. thick and are curved in the shape of an involute, thus providing a constant cooling channel width. The plates are of complex sandwich-type construction composed of uranium oxide–aluminum cermet held between aluminum covers. In addition, the fuel concentration varies along the involute to flatten the power distribution.

The fuel region is surrounded by a concentric ring of beryllium reflector ~1 ft thick. This, in turn, is backed up by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water.

The reactor core assembly is contained in a 8-ft-diameter pressure vessel, which is located in an 18-ft-diameter cylindrical pool of water. The top of the pressure vessel is 17 ft below the pool water level, and the reactor midplane is 27.5 ft below the pool level.

A suggestion by J. A. Lane was the genesis of the HFIR. He proposed, in 1956, that a group of six Oak Ridge School of Reactor Technology students take as a project the design of a flux-trap reactor with emphasis on exploring the potential of the flux trap. The leader of that group was R. D. Cheverton.

Following initial operation of the ORR in December 1957, discussions began at ORNL concerning the need for thermal fluxes an order of magnitude greater than were then available. As a result of the initial meeting, a series of informal seminars was conducted to explore the need further and to examine, in some detail, the technical problems associated with the design and construction of a reactor capable of producing such fluxes. The primary conclusion reached was that the most pressing need for the high thermal-neutron fluxes (i.e., 3 to $5 \times 10^{15}$ neutrons/cm$^2$ s) existed in connection with the production of transuranium elements and other isotopes.

Following a critical review of the status of the transuranium production program by the AEC Division of Research at a meeting in January 1958, it was decided to embark on a program designed to meet the anticipated needs for transuranium isotopes. This was to be done by undertaking irradiations in existing reactors. By late 1958, it became apparent that acceleration of this program was desirable. Therefore, a meeting was held in Washington, D.C., in November 1958. Following this, it was recommended that a high-flux reactor be designed, built, and operated at ORNL, with construction to start in fiscal year 1961.

Consequently, ORNL submitted a proposal to AEC in March 1959. Authorization to proceed with the design of a high-flux reactor was received in July 1959, and a preliminary conceptual design was published in 1959. C. E. Winters was the Project Director at that time. J. A. Lane was a strong proponent of the reactor and was instrumental in securing funding for it.

Development of design criteria for the reactor facility was begun by ORNL in 1959, and by March 1960 a general description of the proposed HFIR was published. The firm of Singmaster and Breyer was retained in March 1960 as architect-engineer for the purpose of handling detailed design of the non-nuclear portions of the plant.

Responsibility for the reactor core and control and safety systems was retained by ORNL.

A. L. Boch replaced C. E. Winters as HFIR Project Director in December 1961. T. E. Cole was appointed Technical Associate Director.

Substantial design, analysis, and test work was conducted in the Reactor Division. The persons involved were from sections headed by M. Bender and M. I. Lundin, S. E. Beall, A. M. Perry, and I. Spiewak. These included T. G. Chapman; H. C. Claiborne; J. W. Hill, Jr.; N. Hilvety; W. H. Kelley; H. A. McClain; J. R. McWherter; R. E. Shappel; J. R. Westsik; and R. S. Valachovic. R. D. Cheverton was responsible for reactor physics and companion analyses, J. E. Jones was responsible for control plate drive development, and W. G. Cobb was design coordinator. Other ORNL personnel involved in the HFIR Project included assignees from the Instrumentation and Controls, Chemical Technology, and Metals and Ceramics Divisions as well as the Inspection Engineering Department.

In early 1965, construction was complete, and final hydraulic and mechanical testing was begun. Criticality was reached on August 5, 1965. Low-power testing was completed in January 1966, and power operation at 20 MW(t) was initiated. Full design power of 100 MW(t) was reached on September 9, 1966. The HFIR has operated successfully since that time and has been an outstanding research tool.

## 4.4 STUDIES AND EVALUATIONS

The purpose of work in this area was to review and evaluate power reactor concepts being studied by AEC. The scope of the studies and evaluations embraced most of the proposed concepts for power generation. The objectives were to conduct reactor analyses and to determine capital, operating, and fuel cycle costs. The latter included fuel fabrication, reprocessing, and refabrication costs. The results were for use by AEC to aid in making decisions regarding concepts to be supported.

These studies and evaluations constituted a continuing, self-contained program within ORNL and spanned the period from about 1955 to 1974. The activity, in the end, evolved into economic studies carried out by H. I. Bowers under I. Spiewak, and these have continued to the present under C. R. Hudson.

J. A. Lane initiated and led the work for a major share of the time. He continued to provide ideas and guidance until 1970.

In the beginning, the work was both carried out and integrated in the Reactor Analysis Section of the Reactor Experimental Engineering Division under P. R. Kasten. Other divisions providing input were Metallurgy (later Metals and Ceramics) and Chemical Technology.

When the Reactor Division was formed in 1960, the work was placed in the Reactor Analysis Section under A. M. Perry; P. R. Kasten was associate head of the section and continued to direct the technical activities until he took a leave of absence from ORNL in 1963.* Following his return in January 1965, he again became involved in the reactor studies and evaluations work. He continued to participate throughout the duration of the program in the economic and technical evaluations of various reactor concepts and assumed J. A. Lane's leadership role for a year in 1967 when Lane was in the Philippine Islands.

M. W. Rosenthal was technical coordinator of studies and evaluations work from 1963 to 1966. R. S. Carlsmith succeeded M. W. Rosenthal as technical coordinator of advanced converter reactor studies; he became Director of Studies and Evaluations in 1969. L. L. Bennett, who was in I. Spiewak's section, followed Carlsmith as Director in 1971. The work, as mentioned earlier,

*He was Guest Director of the Institute of Reactor Development in Jülich, Germany, in 1963 and 1964.

changed form in 1974, with emphasis on establishing uniform ground rules and cost factors for comparative evaluations of nuclear power plants and coal-fired plants. Many other Reactor Division personnel, including E. S. Bettis, H. C. Claiborne, J. G. Delene, E. H. Gift, C. G. Lawson, M. L. Myers, R. C. Olson, J. P. Sanders, J. L. Wantland, J. H. Westsik, and D. R. Vondy, contributed to the work done.

These reactor studies and evaluations were a part of the overall AEC program for developing reactor concepts. During the 1950s a number of projects on reactor concepts were pursued by AEC, including the Homogeneous Reactor Experiment/Test (HRE/HRT) and the MSR at ORNL plus the Experimental Breeder Reactor and the Experimental Boiling Water Reactor (BWR) concepts at Argonne National Laboratory. Among the pilot plants included in this development program were the Shippingport Atomic Power Station [60 MW(e)] in Pennsylvania; an organic-moderated and -cooled reactor plant [11.4 MW(e) net] in Piqua, Ohio; the Hallam Nuclear Power Facility [80 MW(e)] in Nebraska; and the Carolinas-Virginia Tube Reactor Plant [17 MW(e)] in Shoals, South Carolina. The first of these pilot plants had a pressurized-water reactor (PWR),* the third had a sodium-cooled, graphite-moderated reactor, and the last had a heavy-water moderated and cooled reactor.

Reactor evaluations and economic studies were conducted at ORNL in conjunction with this program to make relative quantitative comparisons of the various reactor concepts. In the mid-1960s, the ORNL multidivision reactor evaluation and study activities became a part of a larger AEC multilaboratory, multisubcontractor program for evaluating and comparing reactor types for power generation.

Early in the 1960s, the majority of the reactor concepts under development in the United States for central station power generation were considered to fall into three categories: PWRs (including BWRs), advanced-converter reactors, and breeder reactors. Forecasts at that time postulated that an initial period of PWR and BWR construction would ultimately be followed by an era of breeder reactor dominance. Advanced converters would be used in the intermediate period. A study directed toward determining which converters would be attractive during the period was therefore done. The measure of attractiveness was plant economics. Four advanced converter reactor types were selected for the study; a PWR was included in the evaluation to provide a comparison with advanced converters.

High-temperature, gas-cooled, converter reactors were projected to have the lowest power cost. This cost was ~84% of that projected for a PWR. This was an ORNL study begun in 1963; the report was published in January 1965. M. W. Rosenthal was coordinator of this multidivision study.

Other specific studies and evaluations conducted and the roles of the organizations involved are illustrated by the following. An overall assessment of the AEC's Civilian Nuclear Power Program was initiated at the request of the Joint Committee on Atomic Energy in 1966. Reactor types considered included LWRs; heavy-water-moderated, organic-cooled reactors; HTGRs; heavy-water-moderated, boiling-light-water-cooled reactors; and steam-cooled, fast, breeder reactors. Each evaluation was made on the basis of a 1000-MW(e) plant design. The individual study reports were prepared by AEC-appointed task forces, which included representatives from the reactor industry, national laboratories, and AEC, or by ORNL. Reviews were done by a combination of reactor industry, national laboratory, and AEC personnel.

In the forewords to the reports from these studies, which were published in 1969, M. Shaw, Director, Division of Reactor Development and Technology (DRDT) of AEC, pointed out the following. A reactor concept other than light-water-moderated and-cooled reactors and LMFBRs would have to overcome at least two important factors: the

*This was the first civilian application of a PWR.

widespread acceptance of the LWR and the availability of funds over and above those necessary to meet the commitment to LMFBRs. Subsequently, congressional support was provided to LWR, LMFBRs, and HTGRs, effectively precluding further work on molten salt and other power reactor types.

## 4.5 SPACE-ORIENTED PROGRAMS

### 4.5.1 Medium-Power Reactor Experiment (MPRE)

Space Power Program work on auxiliary power systems was initiated at ORNL in 1958, when two U.S. Air Force representatives, who became disillusioned with work under Systems for Nuclear Auxiliary Power (SNAP) programs, asked A. P. Fraas to produce a more promising concept than was being pursued. Although reluctant because of other commitments, he agreed to identify fruitful avenues for providing power output in the 20- to 300-kW(e) range, which was of prime interest.

A space nuclear power system has the usual components—a reactor heat source, an energy conversion unit, and a means for rejecting heat. The only practical method of rejecting heat is by space radiation. For this, the operating temperature of the reactor must be high enough to achieve reasonable thermodynamic efficiency of energy conversion. Thus, the first step was to choose a thermodynamic cycle and working fluid that would give acceptable size and weight of the radiator for dissipating waste heat because this is the largest and possibly the heaviest component of the system.

The second step was to investigate system reliability. This investigation showed that the requisite reliability could be achieved only by using a single-loop system with components such that a matched set could be integrated to form a system that could be slaved to the load (i.e., power output of the system would automatically correspond to power demand). To keep it sufficiently simple, no valves or electronic equipment were to be used to modulate vapor flow from the nuclear boiler to the turbine or liquid flow between the condenser and boiler. With the help of A. M. Perry, it was concluded that a system using liquid potassium as a coolant and potassium vapor to drive a turbine would meet the requirements established.

A study of the feasibility of potassium use in this application was funded by the ANP Project. Also covered under this funding were more detailed design and performance investigations of key aspects.

By 1961, an encouraging set of results was in hand, and a comprehensive set of development tests could be laid out for a program designed to yield an experimental unit. The first item in the program was an extensive set of detailed reactor physics analyses closely coupled with critical experiments. This work was done under direction of A. M. Perry; it was successful and provided a firm foundation for the program to develop.

The second item was a set of boiling flow and system stability and control experiments. Electric cartridge heaters were used to simulate reactor fuel elements, and water was used as the surrogate for potassium working fluid. Two systems were built; one had a 7-rod cartridge heater, while the second had a 91-rod cartridge heater. R. E. MacPherson, assisted by A. G. Grindell, H. C. Young, D. L. Clark, and others, was responsible for development of the electric heater rods; these same persons and M. M. Yarosh were responsible for the test systems. Operational acceptability of key components under zero-gravity conditions was examined separately.

The boiling flow and system stability and control experiments clearly showed that the system would work. This was a triumph because of the many novel features in the system and prevailing skepticism.

The MPRE assumed program status in 1964 with A. P. Fraas as Director. He reported to H. G. MacPherson, Assistant ORNL Director, and, later, to F. L. Culler when he succeeded H. G. MacPherson.

Design of a full-scale reactor system proceeded concurrently with component tests. G. Samuels, with the aid of M. E. Lackey, R. S. Holcomb, and others, was responsible for the system as a whole; S. E. Moore, G. T. Yahr, and coworkers conducted stress analyses; and F. C. Zapp and others handled the test facility design.

The 91-rod system continued to operate for 2850 h at which point it was shut down because many of the heater rods were no longer functioning. At this point, the director of the AEC Space Power Program ordered termination of the program. The reason given was lack of funding. ORNL protested, but these protests were ineffectual; the program was terminated in 1966.

### 4.5.2 Space Programs

A Space Reactors Office* was established in the Reactor Division in the early 1960s with A. J. Miller as coordinator of work in this area. The sponsor was the AEC Space Nuclear Propulsion Office (SNPO), a joint National Aeronautics and Space Administration (NASA) and AEC office for addressing space applications of nuclear power and other joint interests. The ORNL office dealt with such things as power plant design and performance (Isotope and Reactor Divisions), high-temperature materials for rockets that must withstand high-speed reentry into the earth's atmosphere (Metals and Ceramics Division), and biological studies on the effects of weightlessness (Biology Division).

In particular, the work was addressed to providing electric power aboard space vehicles of relatively long life. For use in a space vehicle, the energy content per pound is very important, thus limiting the power sources of interest to those sources depending on solar radiation, nuclear fission, or radioactive decay. To meet the anticipated need for power supplies in space, a series of SNAP reactors was developed.

The Isotopes Division addressed the development of isotopic heat sources for power plants in space vehicles, while Reactor Division work was in support of nuclear power source (SNAP) design and development activities, which had been ongoing since the 1950s. The latter was conducted by A. P. Fraas, H. W. Savage, R. E. MacPherson, and coworkers. The approximate time span of the ORNL involvement in Space Programs work was from 1962 to 1968. The Office was in the Reactor Division until 1967 when it was placed under F. L. Culler, Assistant ORNL Director.

## 4.6 NUCLEAR SAFETY

Two main areas are addressed under this heading. They are the *Nuclear Safety* journal and the Nuclear Safety, or the Nuclear Safety Research and Development, Program.

### 4.6.1 *Nuclear Safety* Journal

ORNL agreed to assume the editorial responsibility for the journal *Nuclear Safety* in January 1959, and Vol. 1 No. 1 appeared in September of that year. *Nuclear Safety* was initially one of a number of technical journals sponsored by AEC's Technical Information Service. These journals were called "technical progress reviews" because of their contents, or "rainbow series" because of their colors. The series started with *Power Reactor Technology* (green), a quarterly that first appeared in December 1957, and was soon followed by *Reactor Fuel Processing* (purple), starting in February 1958, and *Reactor Materials* (blue), starting in March 1958. *Nuclear Safety* (gold) appeared in September 1959, as stated above, and the last in the series, *Isotopes and Radiation Technology* (purple), started in the fall of 1963. *Nuclear Safety* soon outdistanced all others (in terms of subscriptions) and is the only one that survived.

*The title was later changed to Space Programs Office.

Richard M. Berg, Assistant Chief, Industrial Information Branch of the AEC visited A. M. Weinberg, who was Director of ORNL, in December 1958 to explore possible interest on the part of ORNL in preparing a quarterly technical progress review on nuclear safety. Weinberg subsequently responded by letter stating that ORNL was prepared to undertake the preparation of this review journal.

Before this response, the Reactor Projects Division, then headed by R. A. Charpie, was determined to be the logical choice for organization of the material for *Nuclear Safety*. By mutual consent, the editorial responsibility was assigned to W. B. Cottrell, who was then a group leader in the Reactor Projects Division in charge of reactor hazards analysis.

*Nuclear Safety* reviews the literature and recent developments regarding the safety aspects of reactors and the nuclear fuel cycle, including mining, fuel reprocessing, storage, and shipment. Coverage includes the design, licensing, construction, and operation of reactors and fuel cycle facilities as well as their effects on the environment. Also reviewed and reported on are foreign-reactor safety programs, important safety-related conferences and meetings, current safety research and development, reactor licensing actions, operating experiences, and opinions of the Advisory Committee on Reactor Safeguards. Most articles are in-depth technical reviews of selected topics by nationally and/or internationally recognized authorities and contain extensive references. *Nuclear Safety* is used by reactor designers, builders, and operators and by researchers, administrators, and public-safety officials in both government and private industry.

The first editorial staff of *Nuclear Safety* consisted of W. B. Cottrell (Editor), R. A. Charpie (Advisory Editor), and five assistant editors each of whom was assigned the responsibility for a particular section: L. A. Mann—I. Nuclear Safety Criteria; C. G. Bell—II. Accident Analysis; C. S. Walker—III. Reactor Safety Features; W. B. Cottrell (dual capacity)—IV. Safety Features in Plant Design; L. L. Emerson—V. Activity Release and Consequences; and H. N. Culver (then with ORNL on loan from the TVA)—VI. Recent Developments. In addition, A. W. Savolainen, technical editor, and A. S. Behr, research librarian, supported the staff. Of course, style, format, and modus operandi changed significantly in succeeding years. In the period from 1965 to 1973, *Nuclear Safety* had a managing editor, J. P. Blakely.

### 4.6.2 Nuclear Safety Program

This program at ORNL was established when the scope of earlier safety investigations was enlarged in 1961. F. R. Bruce was the first coordinator of the program; the role of program coordinator was transferred to W. B. Cottrell in 1964. F. R. Bruce was Assistant Deputy Director of ORNL, and W. B. Cottrell was on staff assignment in the Reactor Division.

At that time, the Nuclear Safety Program addressed five areas:

1. obtaining data needed to assess realistically the consequences of accidents in reactors and chemical plants as far as the releases and transport behaviors of fission products are concerned;
2. developing and evaluating countermeasures to be employed in the event of accidents entailing radioactive materials;
3. providing R&D support in the area of fission product sampling devices and interpretation of fission product release and transport phenomena;
4. critically evaluating and compiling, in handbook form, information on reactor containment; and
5. collecting, interpreting, and reporting information in certain key areas of reactor safety for the nuclear safety community.

The Reactor Chemistry, Chemical Technology, and Reactor Divisions were involved in this program. The Reactor Division addressed areas 4 and 5 and participated in addressing areas 1 and 2 through work in the pilot plant described below.

The Nuclear Safety Program was an umbrella program for many smaller, individual programs or projects. Most were funded by the AEC Division of Reactor Development (later, DRDT) and administered by branches reporting to the Assistant Director of Nuclear Safety. For reporting purposes, the work was broadly divided into categories. Both work scopes and categories changed with time, as would be expected. The emphasis here is upon work done by the Reactor Division.

### 4.6.3 Nuclear Safety Pilot Plant (NSPP)

The NSPP was built in the 7500 area, primarily in the chemical processing cell in the HRT building. Operated by Reactor Division personnel, it was used for work in the first two areas listed. The purpose of the NSPP was to evaluate the transport behavior of fission products, or simulated fission products, in a simulated containment system of sufficient size to permit extrapolation of fission product behavior observed in laboratory-scale systems up to actual containment systems. It was not intended to allow simulation of the fission-product-release spectrum of any postulated accident but, rather, to attempt to produce a wide range of fission product types and to observe the behaviors of these under various conditions in confined volumes. Another function was to carry out tests of engineering safeguards, such as filter systems. Further, the NSPP was designed for experiments with fuel containing up to 1000 Ci of mixed fission product activity, and provision was made for remote loading and unloading of fuel specimens and for remote recovery of all sampling devices.

L. F. Parsly, J.K. Franzreb, M. H. Fontana, P. P. Holz , J. L. Wantland, and others were engaged in the NSPP work. NSPP activity areas are described below.

**Spray and Adsorption Technology.** In a loss-of-coolant accident (LOCA) involving a large power reactor, even with effective cooling of both the core and containment building, some release of volatile fission products into the containment building may occur. Water sprays are included in reactor plants as pressure reduction devices; the use of additives to the spray solutions for pressure reduction can also result in a reduction of fission product concentration in a building. Therefore, ORNL conducted a spray technology program to investigate the use of various spray solutions in removing contaminants.

The NSPP played an important role in conducting radioiodine gas removal experiments. The objective was to determine the mechanisms of removal from a containment atmosphere for analytical model development. The spray performance model developed can be used to calculate iodine concentration vs time.

Fission-particle removal experiments were also conducted. In general, the tests demonstrated that these particles were effectively removed by sprays. These studies, which were begun in 1964, were completed in 1970.

**Filtration and Adsorption Technology.** Once activity is released to the atmosphere, as during the melting of a fuel element, the circulation of this contaminated atmosphere through filters is the most effective proven and feasible technique of "fixing" the activity. In addition to the many variables involved in a release, the technology is inherently concerned with reliable maintenance of high-removal efficiencies for all fission products throughout the design life of the filter media. The presence of moisture, such as would be extant in the containment atmosphere following a LWR accident, generally has a detrimental effect on filter performance.

Promising filter combinations were tested in a 20-$ft^3$/min recycle loop in the NSPP under conditions of temperature, pressure, and atmosphere most closely simulating the containment atmosphere in a reactor accident. The loop was placed in service in 1965. The test unit, which was a combined moisture separator and fission-product removal system, included a demister, an absolute filter, and a charcoal bed incorporated into a remotely removable canister.

Results obtained gave no evidence that high-efficiency, particulate filter performance was severely impaired by operation at 100% humidity. They also demonstrated that, when fuel meltdown occurs under reducing conditions (steam and hydrogen present), a large fraction of the iodine is associated with particulate matter and is collected in the absolute filter. When meltdown occurs under oxidizing conditions, practically all the iodine passes through the absolute filter and is collected in the charcoal bed. These studies were completed in 1966.

**Transport of Fission Products in Containment Vessels**. In water-cooled reactors, the LOCA is considered to be the most likely maximum credible accident. Fission-product release immediately after failure of the cladding of Zircaloy-clad uranium oxide fuel elements is of primary interest in evaluating the consequences of a LOCA. It was generally assumed that ECCSs would prevent gross fuel meltdown and that only a small fraction of the elements would rupture and release the gaseous fission products existing in void spaces as a result of normal operation.

The transport of these fission products inside the containment was an important concern. To address this concern, the transport behavior of fission products in closed vessels was studied in the NSPP. In this study, experiments were conducted to collect data for deriving and testing mathematical models of fission gas transport within containment vessels due to natural phenomena. These experiments and the mathematical modeling were done by Reactor Division personnel in the 1967 to 1968 period.

The NSPP was reopened under the direction of M. H. Fontana, T. S. Kress, L. F. Parsley, and R. E. Adams in 1976 to address the issue of sodium fire behavior associated with LMFBRs. This facility was subsequently converted for use in developing and validating aerosol behavior models with severe accidents experienced by LWRs.

### 4.6.4 Reactor Containment Handbook

A reactor containment handbook was prepared by ORNL for the AEC in fulfillment of Area 4 needs, as described; W. B. Cottrell and H. B. Piper were responsible parties. The purpose of the handbook was to provide detailed information useful in the design, construction, testing, and operation of reactor containment systems. Work on the handbook was begun in 1962, and the two-volume document was published in August 1965.

### 4.6.5 Nuclear Safety Information Center

Area 5 was and is addressed by the Nuclear Safety Information Center (NSIC). This center was established in 1963 by the AEC DRDT to collect, assimilate, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. This information is collected from reports and articles, and abstracts and summaries are filed. NSIC also produces reports on safety and other topics as well as the current state of the art in various technical areas.

Abstracts and summaries were initially filed using 5- by 8-in. cards; later, computer files were established and maintained. Also, at the outset, a program of selective, automatic dissemination of information was set up whereby abstract cards were mailed using interest profiles based on information requests received. These mailings were sent out twice each month. An *Indexed*

*Bibliography of Accessions* was published quarterly; it was sorted by NSIC subject categories and had key word and author indexes.

W. B. Cottrell was NSIC Director and J. R. Buchanan was Assistant Director. Original staff members were as follows: K. E. Cowser—Health Physics, C. S. Walker—Instruments and Controls, G. W. Keilholtz—Fission Products, W. K. Ergen—Reactor Transients, F. Gifford—Meteorology, and J. R. Buchanan (dual capacity)—Containment. NSIC grew, and new areas were added; the maximum number was 21. The operations area, which would become the area of concentration, was added after NSIC had been in operation for a short period of time.

### 4.6.6 Pressure Vessel and Piping Technology

Three programs—Pressure Vessel, Piping, and Heavy-Section Steel Technology (HSST)—are included under this topic. The Pressure Vessel (or Experimental and Analytical Investigation of Nozzles) Program was initiated in 1962 but did not come under the purview of the Nuclear Safety Program until 1966. Both the Piping and HSST Programs were initiated in 1967 and were under the purview of the Nuclear Safety Program at the outset. These programs will be discussed in Sects. 4.11 and 4.12.

A review report on the current state of steel pressure vessels for water-cooled reactors was completed under the direction of G. D. Whitman and issued in December of 1967. The objective was to summarize the current state of pressure vessel technology to aid in the assessment of the risk of catastrophic failure of pressure vessels. The study, which included considerations of both boiling- and pressurized-water systems, addressed the major subjects of design, materials, fabrication, and inspection. Worth of upgrading governing codes and standards was considered along with research that would increase knowledge for prediction of vessel quality and behavior during operation.

### 4.6.7 Antiseismic Design of Nuclear Facilities

The proposed location of a nuclear reactor in the vicinity of earthquake faults gave rise to concern that future earth movement might extend to the nuclear site and both cause a major accident and invalidate the engineered safeguard features provided with the plant. To resolve this dilemma, an earthquake program was officially initiated by a letter, dated October 4, 1966, from M. Shaw, Director of the AEC DRDT. This letter requested that ORNL accept the role of central coordinating group for assessment of the significance of possible areas of research, assignment of priority, and evaluation of techniques and results in the development of earthquake-resistant reactor systems.

Three specific tasks were proposed in the letter: (1) development of conceptual antiseismic designs; (2) development of model testing parameters, test methods, and analyses; and (3) interim scale-model screening tests. R. N. Lyon was chosen to lead this activity; G. D. Whitman succeeded him in this role in 1969.

The ORNL program was therefore initially directed to the design problem of differential displacement, to development of more efficient methods for resisting earth shaking, and to accumulating information and methods that might lead to better site selection and better design standards to minimize the contributions of earthquakes to the risk of irradiation exposure from a nuclear facility.

Because the characteristics and load-bearing behaviors of soil at a plant site are important factors in determining earthquake influences on reactor structures, investigations of soil behaviors were conducted. These investigations were done under subcontract and involved (1) in-situ determinations of dynamic soil properties, (2) a study on the liquefaction potential of cohesionless soils, and (3) the development of procedures for predicting the dynamic response of soil deposits.

In a second study, a team of experts from the University of California at Los Angeles carried out two series of vibration tests at the EGCR. The final series culminated in strong excitation of the plant by a 2000-lb charge of dynamite detonated in a shallow drilled hole. Responses of the soil, structure, and equipment demonstrated the importance of this method for both studying strong-motion excitation and verifying plant design.

Agreement could not be reached on an overall program plan, however. ORNL management of the program and subcontract activities were terminated in 1973, and AEC assumed the responsibility for further development and administration of the program.

### 4.6.8 HTGR Safety*

Safety studies addressed specifically to HTGRs were begun in 1966. Initially, the studies were addressed to steam-graphite reactions and fission product behavior. Personnel from both the Reactor Chemistry and the Reactor Division were involved. The steam-graphite reaction studies were done on both small and large scales, with the small-scale work being done on irradiated and nonirradiated specimens. The large- or engineering-scale tests on steam-graphite reactions were conducted by Reactor Division personnel, including F. H. Neill, R. E. Helms, and T. S. Kress.

The fission product behavior studies addressed properties of various chemical forms in the gaseous state and identification of the more important fission products released at high accident temperatures. They were conducted by Reactor Chemistry Division personnel.

Irradiation testing was used to examine high fuel burnup and high-temperature effects on fuel integrity and fission product release. The high burnup testing was done by Reactor Chemistry Division personnel, and the high-temperature testing was done by Reactor Division personnel, including J. A. Conlin and C. L. Segaser.

The engineering-scale tests were terminated in mid-1968 along with the ORNL work on mechanisms of fission product mobility. Emphasis was then given to in-reactor experiments on fission product release and chemical reactions. The resulting tasks were to study in-reactor steam-graphite reactions of fuels and to examine fuel integrity at high temperature.

Fission product distribution in the primary coolant circuit and the effects of steam leakage and depressurization on this distribution were also studied. All work following the change in emphasis was done by Reactor Chemistry Division Personnel. Fission product measurements were taken in the primary coolant and other circuits of the Peach Bottom Reactor as a part of follow-on activities.

### 4.6.9 HTGR Safety Program Office

The HTGR Safety Program Office (SPO) was established at ORNL in April 1966 at the request of the AEC DRDT. The purpose of the SPO was to act as an extension of the DRDT nuclear safety organization in evaluating and coordinating HTGR safety programs being conducted and to provide DRDT with competent and independent recommendations on the direction and implementation of such programs in the future. While the SPO formed a part of the ORNL effort in the nuclear safety field, it reported independently to the Laboratory Director's staff because its mission involved evaluation of various contractor programs on HTGR safety, including that at ORNL.

The SPO was headed by S. I. Kaplan; E. A. Nephew, M. D. Silverman, E. R. Taylor, and others participated in the activities. AEC support for the SPO waned in 1971, and the office was closed out.

*Note that HTGR safety work is also described under GCRP, Advanced Reactor Development, because of its close relationship to that program.

### 4.6.10 Molten Salt Breeder Reactor (MSBR) Safety

Safety studies on the MSBR were conducted in 1967. These were limited to analytical studies of reactor system dynamics, reactor stability under load change conditions, ECCS requirements, and removal of heat generated after reactor shutdown. The MSBR had neither the large stored energy in the high-pressure primary coolant nor the potential of metal-water or steam-graphite reactions associated with gas- and water-cooled reactors. These safety studies were done by MSR Project personnel.

### 4.6.11 Failure Modes of Zircaloy-Clad Fuel Rods

Work in this area was stimulated by concern that the failure mode and behavior of fuel cladding might result in dimensional changes sufficient to alter thermal response of the core to ECCS* operation and, therefore, affect the accident outcome for BWRs and PWRs. Because it was ascertained that the seriousness of failure-mode effects could not be judged because understanding of the events that could occur during the LOCA and the properties and behavior of the fuel rods when subjected to the accident were too limited, failure modes of the fuel rods were of prime concern. Two questions of the following form required answers:

1. Would dimensional changes in fuel rods be of such magnitude and occur to such an extent over the entire core that the efficiency of the ECCS would be significantly impaired?
2. Would the fuel rods retain their integrity on cooling from the accident temperature transient and, if not, how would this affect the heat removal?

Because fuel rod behavior would surely affect the release of the fission products, it was also deemed desirable to determine the fraction of available fission products that would be released from ruptured fuel elements, in what chemical form the products would exist, and what effect the failure mode would have on these.

AEC assigned the responsibility for coordinating all efforts in the study of Zircaloy cladding failure modes to ORNL. P. L. Rittenhouse, Metals and Ceramics Division, coordinated the work.

Areas addressed were (1) high-temperature properties of Zircaloy cladding, (2) cladding behavior in a LOCA environment, (3) transient tests of Zircaloy-clad fuel rods, (4) analytical modeling of Zircaloy cladding deformation, and (5) effect of interactions between fuel, cladding, and fission products on fuel rod failure. Area 2 work was done in the Reactor Division by C. G. Lawson, T. H. Mauney, R. H. Chapman, and others.

The objective of Area 2 work was to investigate the mode of failure of Zircaloy-clad uranium oxide fuel rods in an environment simulating that existing in a water-cooled reactor during and after a postulated LOCA. Work was begun in 1968 and completed in 1971. On the basis of the tests conducted, the results were quite encouraging regarding energy removed from the fuel rods during blowdown. This investigation was the forerunner of the PWR Blowdown Heat Transfer Separate Effects Program and the Multirod Burst Test Program, both of which are discussed in this segment on nuclear safety studies.

### 4.6.12 LMFBR Safety

The LMFBR safety work at ORNL was a part of the AEC Fuel Failure Propagation Program, which was aimed at answering a question central to LMFBR safety: Do small failures propagate from fuel pin to fuel pin and across subassemblies of fuel elements, thereby causing extensive damage to the reactor core? This national effort was coordinated by the Argonne National Laboratory.

*The ECCS is a safety system.

The ORNL work was conducted in the Reactor Division by M. H. Fontana (Project Director), R. E. MacPherson, P. G. Gnadt, J. O. Kolb, J. L. Wantland, L. F. Parsly, W. R. Gambill, A. G. Grindell, R. D. Stulting, T. S. Kress, D. G. Thomas, D. L. Clark, B. H. Montgomery, R. H. Morris, and others. Instrumentation and Controls Division and Metals and Ceramics Division personnel also had important input into the project. This ORNL work concentrated initially on those aspects of partial blockage of coolant (sodium) flow through the reactor core that would lead to fuel failure and failure propagation. A failed fuel mockup (FFM) facility was constructed to experimentally examine flow blockage effects. It was a high-power-level, flexible sodium loop with the capability to perform out-of-reactor experiments with simulated LMFBR fuel rods. The fuel rods were simulated by using internally heated electrical heater rods that could be assembled into a 19-rod cluster contained in a hexagonal can. The power input in this case was 1.4 MW. The cluster was later expanded to 61 rods with a power input of 4.5 MW.

The objectives were later broadened to include the investigation of flow and temperature distributions within fuel rod bundles with and without partial blockages to determine the potential for fuel failure and failure propagation. Also included were the detection of events that could lead to failure and the extrapolation of experimental results to predict the behavior of a full-size reactor under hypothetical conditions.

The FFM was designed to provide for a series of seven tests related to flow blockage. The program was initiated in 1969; construction was completed on the FFM in 1970, and the facility was completely checked out.

Analytical capability for treating flow and temperature distributions in the FFM was developed, and studies in support of the FFM program were carried out. This analytical capability was to aid in planning the experimental program and to ensure that the maximum information would be obtained from each experiment.

A dynamically scaled water mockup of the FFM was also designed and fabricated. Through use of this mockup, flow characteristics in the rod bundle could be examined.

Development and refinement of heaters for the FFM was an important adjunct activity. The purpose was to simulate LMFBR fuel assemblies for use over a wide range of conditions up to and including sodium boiling and subsequent absence of sodium due to boiling off. Work in this area was carried out by R. E. MacPherson, R. W. McCulloch, and others.

The first series of experiments in the FFM was intended to investigate the effects of inlet flow blockage, while the second series was to be used to study in-core blockage. However, the program was altered in 1971 to obtain, as soon as possible, the temperature distributions within the rod bundle and the hexagonal can wall for the unblocked bundle. This information was for use in connection with the Fast Flux Test Facility (FFTF), which made use of a liquid-metal-cooled reactor, at Hanford, Washington. In addition to temperature distributions, local internal flow and heat transfer were also to be investigated. The test data were submitted in 1972.

Cooperation with Westinghouse Electric Company in FFTF matters continued. Cooperation was also extended to Westinghouse to aid in Clinch River Breeder Reactor work as well as to others engaged in LMFBR design and in R&D activities.

### 4.6.13 ECCS Hearings

The ECCS hearings were conducted by the Atomic Safety and Licensing Board of AEC for the Division of Regulation and were held in Washington, D.C. The purpose was to hold rule-making hearings on ECCSs of U.S. LWR, civilian, power plants. They were called because of controversy regarding the capability of the ECCS in the event of a major pipe break in the primary system of a PWR or BWR power plant. Testimony was taken from January 18 through August 5, 1972.

The major complainant was the Union of Concerned Scientists; it had focused on the Pilgrim Nuclear Plant to be built by Boston Edison 4 miles southeast of Plymouth, Massachusetts. Because the Union of Concerned Scientists was unable to obtain substantive information regarding plant safety from AEC, Boston Edison, or General Electric, the reactor system designer and manufacturer, these hearings were convened.

ORNL personnel who testified during the hearings were W. B. Cottrell and C. G. Lawson from the Reactor Division and D. O. Hobson and P. L. Rittenhouse from the Metals and Ceramics Division. D. B. Trauger administered the group. In addition, technical reports on subjects germane to the hearings were prepared by W. C. Gambill, J. J. Keyes, and T. S. Kress. Based on the ORNL study, "Failure Modes of Zircaloy-Clad Fuel Rods," the following salient points were made during the hearings by ORNL participants.

1. Phenomenological heat transfer information and pressure and temperature condition data during LOCA transients were insufficient for making predictions.
2. A primary issue was that when Zircaloy reaches a temperature of 2100 to 2300°F in the presence of steam, the steam-Zircaloy reaction becomes catalytic and impossible to control.
3. The information available was not sufficient to ensure that the cladding temperature would not exceed these values.
4. If the Zircaloy fuel cladding temperature exceeded the 2100 to 2300°F range, melt-through of the cladding would occur.

The outcome of these hearings was a new set of rules that (1) more clearly defined some of the heat transfer calculations to be used in analyzing steam cooling of the core; (2) set forth new methods for calculating the amount of embrittlement that would occur in the Zircaloy cladding, affecting the allowable time at temperature; (3) defined requirements for flow blockage calculations; and (4) gave a number of requirements to be met in calculating the consequences of a LOCA (e.g., allowable power profiles in the reactor, etc.). A major change was the resultant use of a new fuel design with smaller diameter fuel rods for more efficient cooling during normal operation and under accident conditions. These changes resulted in longer fuel life and, consequently, improved economics for reactor cores.

An immediate reaction to the ECCS hearings was the cessation of orders for nuclear power plants.

### 4.6.14 ORNL PWR Blowdown Heat Transfer Separate Effects Program

This program addressed heat transfer mechanisms and temperature conditions associated with fuel elements in a reactor core under coolant-pipe rupture accident, or LOCA, conditions. The immediate result of a severe LOCA in a PWR would be the event known as "blowdown," the rapid loss of pressure and water coolant in the reactor vessel as the water turns to steam. The drop in coolant density that would result under blowdown conditions would slow the rate of heat transfer from the fuel rods in the reactor core, permitting a marked rise in their temperature. Shortly thereafter, the reactor's ECCS, activated by loss of pressure, would bring the overheating core under control.

The program focused on pressure, temperature, and coolant flow conditions beginning at the time of the coolant line rupture and continuing to the end of the depressurization or blowdown. A separate effects approach was used wherein the rôle of each primary variable (flow, pressure, power) on transient heat transfer within a reactor core was established independently and then combined to predict the course of a blowdown transient.

A test facility, the Thermal Hydraulic Test Facility (THTF), was built and operated under this program. This was a non-nuclear pressurized-water loop in which the nuclear fuel was simulated by electrically heated rods of the same size, shape, and power capability as actual PWR fuel rods. The original THTF, designed in 1971, was to permit testing at steady-state conditions with high-pressure, high-temperature water in single-tube test sections with either a single heater rod or small heater rod bundles up to 8 ft in length.

In January 1972, plans for modifying the THTF, at that time incomplete, to permit transient and blowdown heat transfer tests on full-length (12-ft) heater rods in a 7 by 7 array were begun under the leadership of H. W. Hoffman, C. G. Lawson, and R. H. Chapman. Mechanical and electrical design, under the overall supervision of M. I. Lundin and L. V. Wilson, was carried out by J. L. Crowley, C. J. Claffey, C. W. Collins, W. K. Furlong, H. R. Payne, F. C. Zapp, W. M. Brown, R. D. Stulting, and others. Members of the Instrumentation and Controls Division developed specifications for the data acquisition system and designed instruments and controls. THTF modification and operation, heater rod procurement, and bundle assembly were under the supervision of R. E. MacPherson. R. E. Helms, THTF Project Engineer, supervised THTF modifications. Heater rod development was initially under the direction of D. L. Clark and, subsequently, R. W. McCulloch. Bundle 1 was assembled under the direction of A. M. Smith. The first isothermal blowdown test was conducted in February 1975.

This program became a part of the overall LWR safety research program of the NRC. Other parts of the NRC program covered a wide range of experimental and analytical efforts from laboratory-scale experiments to small-scale experimental nuclear plants. Separate-effects studies fell in scale between laboratory-scale experiments and small-scale nuclear plants and were designed to answer specific questions relevant to the hypothetical LOCA. All separate-effects studies were intimately related to large integral studies as well as to analysis programs for developing improved computer codes to model both components and systems.

The THTF was an all stainless steel recirculating loop fabricated from 4-in. pipe. The principal components included a test section (~19 ft of 10-in. pipe and flanges), three heat exchangers with a total heat removal capacity of 7.5 MW, and a pump that developed a pressure rise of 840 psi at a flow rate of 700 gal/min.

The rod bundles tested consisted of forty-nine 0.422-in.-diameter electrically heated rods spaced on 0.563-in. centers contained in a 4-in. shroud box. Although the heater rod length varied from 18.5 to 21.5 ft, the active heated section consisted of 12 ft of Inconel 600 or cupronickel heating elements on the interior of the rod. The axial power distribution closely matched that typical of a nuclear fuel rod.

The THTF tests were aimed at answering two questions. First, how do changes in variables such as power level, coolant flow rate, and coolant temperature affect the events that follow the simulated accident? And second, how well are these events predicted by computer codes used to evaluate the safety of PWRs? Particular concerns were with learning how much time elapses before critical heat flux* is reached, and exactly how hot the fuel rod simulators become before the emergency core cooling water reaches them.

Therefore, specific objectives were to concurrently determine heat transfer related quantities, including the critical heat flux, time to critical heat flux, and local fluid properties, and to test the ability to predict the behavior of single-rod and 49-rod loops under blowdown conditions. Secondary objectives included obtaining critical heat flux data over a range of steady-state conditions

*Critical heat flux is the point at which cooling capacity for the core is diminished. More technically, it is the heat flow per unit area at which the resistance to heat flow increases due to incipient vapor blanketing of the rod surface.

appropriate to PWRs and evaluating the behavior of test loops during simulated operational upset conditions other than those of a LOCA.

This very intensive, high-profile program was completed in 1982. It produced volumes of quality data and results from many detailed analyses associated with LOCA and other operational upset conditions. An outcome was to be improved rules for analyzing postulated operational upset events.

H. W. Hoffman was Program Manager at the outset. He was succeeded in 1974 by D. G. Thomas. Others engaged in the program during the period up to the end of 1975 included R. H. Chapman, J. L. Crowley, C. G. Lawson, R. F. Bennett, J. D. Sheppard, J. D. White, R. A. Hedrick, L. J. Ott, K. G. Turnage, M. D. White, and M. C. Wynn.

### 4.6.15 Multirod Burst Test Program

During a LOCA, the flow rate of the core coolant decreases, causing the rate of heat transfer to drop dramatically. The temperature of the fuel cladding begins to rise. At the same time, the pressure around the fuel rods decreases, while the pressure within the fuel rods, which builds up from accumulating gaseous fission products during normal operation, is increasing. At some point, fuel rod internal pressure exceeds the external pressure. This pressure excess within the rods can produce bulging and failure of the fuel cladding, a phenomenon that is temperature and pressure dependent. The resulting deformation can, in turn, produce flow blockage in the fuel bundle and inhibit coolant flow supplied by the ECCS, leading to faster and more severe deterioration of the reactor core. Therefore, the Multirod Burst Test Program was established to examine the influence of this pressure imbalance on cladding behavior.

The Multirod Burst Test Program was initiated in July 1974 as an experimental study to (1) delineate the deformation behavior of unirradiated Zircaloy cladding under conditions postulated for a large-break LOCA and (2) provide a data base for assessing the magnitude and distribution of geometric changes in fuel rod cladding in a multirod array and the extent of flow channel restriction that might result. The specimens used for simulating fuel rods were pressurized tubes with internal electric heaters, and during testing they were subjected to steam cooling. The tubes had an 0.422-in. outside diameter and were 0.025 in. thick. Data were obtained from single-rod and multirod experiments. The tests were designed to reveal possible effects of rod-to-rod interactions on ballooning and rupture behavior over a wide range of conditions.

Approximately 110 single-rod tests were conducted to obtain burst pressure differential and deformation vs temperature data for correlations. In later tests, each rod was placed inside a circular tube, or shroud, for testing. The internal pressure range was from 115 to 1335 psi, and the temperature ranged from 2140°F down to 1295°F in these tests.

Six multirod bundle tests were performed on 4 by 4 arrays of rods. One test was conducted on a 6 by 6 array, and two 8 by 8 arrays were tested. Burst temperature and pressure data were obtained in all cases. Both the single-rod and bundle tests showed that local temperature gradients have a marked effect on the deformation behavior of Zircaloy tubes; the more uniform the temperature distribution, the greater (and more uniform) is the deformation. It was concluded that deformation depends not only on inherent metallurgical properties of Zircaloy, but also on rod-to-rod mechanical interactions and all factors that determine the temperature gradients.

Axial, preburst and postburst, pressure profiles were taken for the arrays, and the results were used in making pressure drop (flow resistance) determinations. This evaluation indicated that complete axial flow blockage was highly improbable.

Comparisons of results from single-rod and multirod tests were used to examine influence of adjacent rods on each other's deformation behavior, and how rods swell and burst in relation to heating rate, temperature, and pressure. A further use of results was to provide a data base for computer program development by others to predict behavior of the cladding during a hypothetical LOCA.

R. H. Chapman was Project Manager over the full time period covered (1974 through part of 1982). Additional participants included J. L. Crowley, A. W. Longest, R. W. McCulloch, D. L. Clark, and others. Personnel from the Instrumentation and Controls and the Metals and Ceramics Divisions also contributed to the successful outcome of this program.

#### 4.6.16 Epilogue

The LMFBR safety work, which was actually a part of the ORNL LMFBR Program, was not reported under the Nuclear Safety Program after 1972. In 1973, the Nuclear Safety Program was restructured, and G. G. Fee was appointed Director; he was on the staff of D. B. Trauger, Associate ORNL Director. Program sponsorship was shifted to the newly formed NRC in 1974; also, in that year, G. G. Fee became Reactor Division Director and retained his Program Director position.

### 4.7 NUCLEAR DESALINATION PROGRAM

By 1964, R. P. Hammond, an ORNL staff member and a proponent of large nuclear plants, strongly believed that the larger a nuclear power reactor complex, the more economic a source it is for producing electricity and heat, which could be used for desalting water and supplying thermal energy to other processes, both agricultural and industrial. Hammond's opinion was shared by others, and his ideas received international attention, as the United States, the Soviet Union, Israel, and Mexico began planning desalting plants. The Interior Department's OSW and AEC, at that time, supported a growing program at ORNL on developing large reactors and evaporators and also on basic water research.

Early in 1965 AEC accelerated its program on general technical evaluations of application of nuclear reactors to the desalting of seawater and brackish waters. AEC designated ORNL as its primary technical support organization in this field. Also OSW and AEC entered into an interagency agreement under which ORNL would perform, for OSW, selected development and engineering activities in the field of desalting technology. In the cooperative AEC-OSW program, ORNL would be supported by the OSW in work on improved seawater distillation processes, equipment, and plant designs and by AEC in work on nuclear applications and related technology.

R. P. Hammond was appointed Director of the Nuclear Desalination Program, which was a part of the ORNL Desalination Program under G. Young. The latter was also Assistant ORNL Director. In 1966, the Desalination Program, with G. Young as Director, was placed under F. L. Culler, Assistant ORNL Director. Beginning in 1970, R. P. Hammond reported first to F. L. Culler, then to D.B. Trauger. The latter became Associate ORNL Director when F. L. Culler became ORNL Deputy Director in that year. I. Spiewak was Deputy and Associate Director of the Nuclear Desalination Program until 1973 when he became Director. C. C. Burwell was Associate Director responsible for the AEC-sponsored portion of the program; the OSW-related work was under I. Spiewak. T. D. Anderson replaced C. C. Burwell in 1968 when the latter was assigned to a companion project, the Middle East Study.

General objectives of the AEC Nuclear Desalination Program were to explore the possible applications of nuclear energy to water desalting and other process uses and to cooperate with industry and other government agencies in developing the technical understanding and the hardware to implement desalting applications. The

ORNL program authorized by AEC encompassed assigned tasks in eight areas.

1. Investigate the applicability of reactors being developed under the Civilian Power Program, including the technical feasibility and economic advantages of building very large reactors to supply energy for dual-purpose desalting plants of industrial and agro-industrial complexes* (Nuplexes).
2. Evaluate various types of reactors and reactor concepts for application in meeting the special needs of single-purpose desalting plants.
3. Analyze the coupling of the desalting plant and the electric power generator to the steam supply and to each other and determine methods of achieving operational flexibility to respond to various water and power demands.
4. Determine the problems of dual-purpose plant control and develop control system designs for maintaining stable operation at the desired levels.
5. Investigate the special siting and safety problems of nuclear-powered desalting plants and determine the economic effects of various choices.
6. Evaluate process applications of nuclear energy and study the concept of energy centers as a means of providing food, industrial products, and opportunities for raising the standard of living.
7. Provide technical support for and participation in U.S. and international feasibility studies on the application of nuclear energy to potential industrial or desalination projects.
8. Operate a Nuclear Desalination Information Center.

To assess the application of nuclear energy to both near-term and long-term requirements for desalting, it was necessary to understand interrelationships between the important cost and design parameters of the nuclear energy source, the power station, and seawater evaporator. The most significant parameters were as follows.

1. The use of a single source to produce large quantities of both water and power requires larger reactors than would be needed for power production alone; thus, it was important to evaluate the relative economic potential and the feasibility of extrapolating various reactor concepts to large size.
2. Because different reactor concepts have different capital and operating costs, they could have different potentials for desalting relative to power production.
3. The amount of power produced in a dual-purpose plant depends upon steam conditions. Because each reactor concept produces steam at unique conditions, each concept would produce different ratios of water and power, and their applicability to desalting could be affected.
4. Municipal and industrial requirements in the United States for power and water were such that they could be met with dual-purpose plants. However, for many areas of the world, as well as for agricultural application in the United States, water was needed without large quantities of power that would be produced in dual-purpose plants. Thus, evaluation of process heat reactors without power production was required for this application.

These aspects were addressed in the program pursued.

Studies of reactors with potential for producing low-cost heat for desalting plants were conducted by T. D. Anderson, J. E. Jones, F. G. Welfare, and others. These studies showed that uranium-metal-fueled PWRs offer significant potential for applications including electric power production and dual-purpose desalting. These studies also gave indications that sodium-cooled breeder reactors with unclad metal fuel would yield significant reductions in desalted water cost, making water costs from single-purpose desalting plants competitive with those from dual-purpose power-water

*Complexes that produce water, cheap electrical power, fertilizer, and other products.

desalting plants and thus eliminate the interdependence of power and water production.

Coupling of a desalting plant and an electric power generator to the steam supply and to each other requires unique matching of the prime heat source, the turbine-generator plant for electricity, and the desalination module for the most economical production of the desired quantities of electricity and water, or process heat, electricity, and water. Therefore, this coupling is an extremely important element in a multipurpose complex; coupling studies were a continuing ingredient in the Desalination Program. These studies were led initially by J. C. Moyers, followed by J. E. Jones.

Dual-purpose plant control studies were conducted to develop practical methods for successfully addressing combined-plant control and safety problems. A primary effort was that of developing methods to accurately predict desalination module dynamics because this was the weakest link in addressing overall plant control. In this regard, maintaining distillation-type desalination-module operational stability to avoid brine contamination of the product was identified as a major factor in successful plant operation. In addition, it was found that the closer the coupling scheme approached the economically ideal, fully dual-purpose plant design, the greater the tendency for instabilities to occur. Plant control and related studies were conducted by S. J. Ball (Instrumentation and Controls Division) and J. G. Delene (Reactor Division) plus others from the two divisions.

Siting and safety aspects of large plants were examined by R. C. Olson, W. H. Kelley, and others. Both off-shore and near-load siting were considered.

When overall plant studies indicated that electrical power output of a dual-purpose (electricity and water-producing) plant might be used economically to produce fertilizer and other products usually needed in nonindustrialized areas, the idea of locating an industrial complex near the dual-purpose plant appeared especially promising for application in developing countries. The concept developed rapidly around the central idea of a nuclear desalting complex located at a coastal site in a desert region, where the product water could be used for intensive agricultural production supported by fertilizer and perhaps other products from the industrial complex. Distilled water for irrigation, the best fertilizer scientifically applied, improved crop strains for higher production, and intensive year-round farming could be combined to achieve maximum crop yields in a "food factory." This concept would become the centerpiece for the program.

Study and development of the idea was a major activity in 1967. Professor Edward A. Mason of the Massachusetts Institute of Technology organized a group to do a "summer study" at ORNL to investigate how and to what extent the low-cost energy anticipated from nuclear reactors could be used effectively to increase both industrial and agricultural production, with particular attention being given to applications in developing countries. The study group consisted of 17 full-time scientists and engineers, 6 full- and 13 part-time consultants, 9 participating industrial organizations, and a large number of individual contributors.

The resultant study, embracing industrial and agro-industrial complexes, showed that the "complex" idea was reasonable and might be attractive to developing countries. Returns on investment, when the complex included food production based on the use of desalted water in an arid region, appeared as attractive as the production of more conventional industrial chemicals and metals.

More follow-on detailed elucidation of production capabilities stimulated interest in studies of applications to problems in specific regions. Therefore, studies pertaining to the needs of India, of the southwestern U.S.-Mexico border region, and of Puerto Rico were undertaken. The Middle East Study, described separately, was also initiated as a major effort. J. W. Michel was manager of the India study and participated in other nuclear complex application studies.

Work subsequent to the intensive, multidisciplinary study in 1967 on nuclear-powered industrial and agro-industrial complexes addressed several areas. It embraced evaluations of many energy-consuming industrial processes suitable for nuclear energy complexes, studies of crops and farming techniques, and investigation of subsurface irrigation to develop a system offering a low-cost means for achieving a large increase in water use efficiency. At the same time, a reanalysis of the economic viability of desalted water for agriculture led to the conclusion that the likelihood for the cost of desalted water becoming low enough for use in conventional agriculture was remote, while its use in unconventional agriculture, as envisioned in the food factory concept, appeared promising.

Studies related to the application of nuclear desalting plants to long-term water supply problems included a review of water resources and alternatives for the California region and participation in a study of supplemental water resources for the New York metropolitan region. The New York study was a joint effort by AEC, OSW, the City of New York, the State of New York, and the Consolidated Edison Company.

During the course of desalting studies, a number of desalting processes were considered, including distillation (evaporative), membrane,* and freezing. Major emphasis was given to two distillation processes, the multistage flash (MSF) and the vertical tube evaporator (VTE). Combined VTE-MSF plant designs were also considered.

Those involved in desalting process work, in addition to I. Spiewak, were L. G. Alexander, D. M. Eissenberg, and H. W. Hoffman (improved heat transfer); J. W. Hill, E. C. Hise, and R. Van Winkle (VTE design, flow testing, and pilot plant operation); R. P. Wichner (MSF flow analysis); and D. G. Thomas (membrane process studies). Considering the total program, W. E. Thompson was responsible for administration and reports; information center responsibilities were addressed by K. O. Johnson (Technical Publications Division). Others who contributed included H. I. Bowers, L. D. Chapman, R. H. Chapman, R. S. Carlsmith, C. J. Claffey, T. E. Cole, C. W. Collins, L. C. Fuller, U. C. Fulmer, B. C. Garrett, E. H. Gift, P. P. Holz, L. Jung, L. R. Koffman, J. O. Kolb, R. B. Korsmeyer, J. A. Lane, M. I. Lundin, R. L. Miller, M. L. Myers, H. R. Payne, A. M. Perry, C. M. Podeweltz, S. A. Reed, J. P. Sanders, R. K. Sood, J. V. Wilson, and M. M. Yarosh. Personnel from the ORNL Reactor Chemistry, Metals and Ceramics, Chemical Technology, and General Engineering and Construction Divisions and the K-25 Engineering and Operations Analyses Divisions also participated in the program.

As stated in the *ORNL Review*, the desalination program began to decline and almost totally evaporated in the 1970s as the nation turned to the more critical problems created by the energy crisis. From 1973 to 1985, S. A. Reed and others worked on a project being carried out in Israel—the Multi-Effect Low-Temperature Seawater Desalination Plant at Ashod, Israel. This was a jointly sponsored project with the State Department's AID and OSW being the active U.S. agencies. During this same period, S. A. Reed also managed subcontracts for conducting economic studies of desalting processes.

## 4.8 MIDDLE EAST STUDY

Senator Howard Baker (Tennessee) sponsored Senate Resolution 155 in 1967, passed by the Senate, calling for exploration of the possibility of building nuclear-powered agro-industrial complexes in the Middle East. This action was a

*In the membrane process, osmosis or diffusion, which proceeds through a semipermeable membrane separating two miscible solutions, is the operating mechanism for extracting water from brine solution.

consequence of the "summer study" led by Professor E. A. Mason on industrial and agro-industrial complexes. The report from that study suggested that agro-industrial complexes would be profitable for some developing countries even with near-term technology and indicated the desirability of carrying out studies in greater depth for specific regions with local conditions and resources taken into account.

Resolution 155 recommended an examination of the concept of large water-producing energy centers in areas of the Middle East as a means for providing (1) new jobs for refugees, (2) an increase in agricultural productivity of existing wastelands, (3) a broad base for cooperation between Arab and Israeli governments, and (4) a further demonstration of the U.S. efforts to find peaceful solutions to areas of conflict. Water and war were recognized to be intertwined.

This study was therefore initiated in June 1968 to explore the technical and economic feasibility of using nuclear-powered, dual-purpose plants to provide large amounts of fresh water and electricity for the development of arid regions in the Middle East. The general scope of the Middle East Study included eight areas:

1. collecting, organizing, and analyzing information on the resources and requirements of the Middle East (e.g., water, raw materials, land, fuel, power, people, and national economics);
2. generating new information (building blocks) on Middle East agricultural products, food processing, and peripheral industries to supplement information available from the 1967 energy-center study;
3. selecting possible location sites for agro-industrial complexes;
4. analyzing markets to assist in selection of product mixes to maximize socioeconomic benefits;
5. estimating capital and operating costs for selected agro-industrial complex designs;
6. evaluating socioeconomic benefits of the selected complexes;
7. defining the nature of and the need for initial experimental or pilot projects to help ensure the success of subsequent larger projects; and
8. defining requirements for project implementation, including consideration of education and training, project planning, financing, and management.

The study involved evaluation of national needs for power and water in five Middle East countries (Egypt, Israel, Jordan, Lebanon, and Syria), evaluation of alternatives to nuclear energy in meeting those needs, and design studies and economic analyses of large water-producing energy centers for selected locations.

M. C. Edlund was the first Director; he was on the staff of F. L. Culler, who was Assistant ORNL Director. Edlund left ORNL early in the project life, and J. A. Lane became Director. C. C. Burwell was Associate Director of the project. The technical staff was composed of ORNL personnel and consultants from cooperating government agencies and universities, assisted by an advisory panel. There were seven consultants and eleven advisory panel members. M. C. Edlund became a consultant after he left the project and is included in the total.

Several studies on related topics were carried out. These included markets and potentials for agricultural output, aquaculture,* protein requirements for nutritional planning, nutrition economics, evaporators for desalting, optimization of water storage reservoirs, and export market studies for possible Middle East products. Those contributing to these and other studies included E. C. Hise, D. B. Lloyd, J. C. Moyers, S. A. Thompson, W. C. Yee, and others.

Overall studies addressed to individual countries, that is, Egypt and Israel, and a Middle East regional study were conducted. Agro-industrial

*Aquaculture is the rearing of organisms in an aqueous environment under controlled conditions using the techniques of agriculture and animal husbandry.

complexes were found not to be economically favorable for individual countries. A regional approach, on the other hand, was found to be much more attractive because of the impact on existing problems, which were providing jobs for refugees, food for inhabitants, and a basis for cooperation. Work on the project was completed in 1970, and a briefing was given in Washington, D.C. However, no action was taken on the results obtained.

## 4.9 TERRESTRIAL AND UNDERSEAS POWER PROGRAMS

These programs were established to assist the DRDT of the AEC. G. Samuels was the leader of both programs, with assistance from R. S. Holcomb, M. E. Lackey, and others. The first, which was initiated in 1966, was in support of the Army, while the second was addressed to Navy needs. The activities under the Terrestrial Power, or Terrestrial Low-Power Reactor, Program were in fact extensions of those under the Army Package Power Reactor (APPR) Program.

### 4.9.1 Terrestrial Power Program

The studies conducted were oriented toward conventional power plant design based on updated technology and sufficient redesign to incorporate needed changes identified through operating experiences with APPR, or portable reactor system (PM), plants. An objective was to also carry out necessary development and testing to ensure that a field-erected plant could be operated without the usual growing pains experienced by plants of this type. The target plant, in this case, was to have an electrical output of 1.5 MW and an export steam capacity of 2500 kW for space heating and other needs.

Experience had shown that a major difficulty with the PM-1 near Sundance, Wyoming; the PM-2A in Greenland; and the PM-3A at McMurdo Sound in Antarctica was a lack of working space and access to equipment. The most common complaint of operating crews of those small reactors, in addition to the lack of access to equipment and inadequate work area for performing maintenance, was the general problem of valves. The latter included inferior quality, obsolescence, valves from too many manufacturers, and leakage (especially from relief valves). Most of these could be addressed by better selection and procurement practices. Excessive leakage from the relief valves of PM-1 was corrected by installing rupture disks.

Although the PM-1 and the PM-3A achieved very good operating records, the early experience with these plants was quite discouraging. Both plants were built on very tight schedules that did not allow sufficient time for development or testing. This translated into low plant availability during the first years.

Both BWR and PWR plants were investigated in these studies; again, a PWR plant was selected. The reactor core, in this case, emulated that of a Westinghouse Electric Company PWR.

The plant was to be transportable by surface transportation to its site, and the sizes and weights of the components or modules were not required to be restricted to those suitable for air transport. The plant was to be assembled and operated in support of a power-consuming installation at a remote land-based plant, and it was not intended to be movable after it had been in operation. Field fabrication was to be minimized and site work essentially limited to installation and hook up of equipment. The site or application was to be such that containment might be necessary.

The containment system selected consisted of two vertical, interconnected vessels, each 11 ft in diameter and 35 ft high. One tank, the reactor tank, was to be filled with water to a depth of 25 ft and contained the reactor, spent-fuel storage cask, and a decay heat cooler. The equipment tank was to be dry and to contain only the steam generator, primary coolant pump, pressurizer, and interconnecting piping. This arrangement left the auxiliary equipment, piping, and valves external to the tanks and accessible for maintenance during operation.

In the end, program changes limited accomplishment of the scope of work originally planned. A report covering the reactor plant study was published in August 1969.

### 4.9.2 Underseas Power Program

This program, also known as the Isotope Kilowatt Program, addressed the development of a power system using heat produced by radioactive isotopes. The underlying principle was the use of heat output from the isotopes to heat a dissimilar metal junction, thus converting heat to thermoelectric power (electricity); the mechanism for this conversion is the Seebeck effect.

The components were a heat-block shield assembly that contained the heat-producing isotopes, heat pipes to carry heat to the thermoelectric units, and the units themselves. The heat pipes were straight tubes that provided natural circulation flow of potassium vapor from the heat source to the thermoelectric units as well as providing for return of the potassium liquid by capillary action.

Isotope Division personnel assisted in this project by developing a detailed design for the heat source. A thermoelectric generator was purchased and tested in the Reactor Division, and a task for detailed investigation and development of exterior insulation for the heat-block shield unit was executed.

The heat-block shield unit, heat pipes, and thermoelectric modules were to be contained within a pressure vessel. The space between the cylindrical heat-block shield and the encapsulating pressure vessel was to be filled with insulation. One of the possible accidents for the total assembly was a loss of heat pipe cooling of the heat block because of flow stoppage or loss of fluid from a break in the system. In such an event, the temperature of the heat block would rise, leading to possible melting of the fuel cladding.

To prevent this occurrence, insulation for use around the cylindrical surface of the heat-block shield was designed to also serve as a thermal fuse. In this role, the insulation would melt at a temperature sufficiently low so that the fuel cladding would not exceed a safe temperature well below the melting point. Operationally, the heat-block temperature would increase until the insulation began to melt, and the temperature would level off. The insulation would continue melting until it was reduced to a thickness that would allow transmittal of the full-power heat load at the melting point of the insulation.

Evaluation of materials that might serve as thermal insulation at operating temperature, but would melt at somewhat higher temperature, was chosen as a primary task. The maximum allowable temperature of the fuel cladding was selected as 2000°F for a LOCA. On the basis of this constraint, an insulation, made up of aluminum screen and solid sheets of aluminum foil and placed at every eighth layer of screen, was developed. Expanded-metal, stainless-steel mesh was placed at the inner and outer surfaces of the insulation matrix to serve as support for the assembly.

Tests of this insulation led to the following conclusions. The aluminum insulation will melt and limit the maximum fuel cladding temperature to ~2090°F with the heat block (made of carbon steel) vertical, or ~2150°F with the heat block horizontal, when argon or nitrogen gas is used in the heat-block shield pressure vessel. For a nickel heat block, these temperatures would be 200°F lower.

A report on the insulation studies was published in August 1973. The combined results from the total project showed that an isotopic power source could be used to both supply power and maintain integrity in accordance with international law.

## 4.10 HOUSING AND URBAN DEVELOPMENT PROGRAM

In the late 1960s there were rumors of an impending oil shortage, and interest in nuclear power plants was high. However, nuclear power plants were operating at lower efficiencies than fossil-fueled plants, thereby releasing more waste heat to water, including both streams and lakes, or directly to the atmosphere. These releases were recognized as environmentally undesirable.

S. E. Beall suggested that uses for this waste heat be examined by A. J. Miller and others in the Reactor Division. At the time that suggestion was made, the Department of Housing and Urban Development (HUD), in keeping with the "Great Society" years under President L. B. Johnson, was examining the feasibility of establishing about a dozen new cities in the United States that would be around 100,000 to 2,000,000 in population. Therefore, A. J. Miller, J. T. Meador, and others undertook a HUD-sponsored study on an energy system for a city with 480,000 people; A. J. Miller was the ORNL HUD Program Coordinator. Nuclear reactors and steam turbines were to be used for supplying electricity along with chilled water for air conditioning and 400°F water for such things as industrial use, district heating, and greenhouse heating.

When the city study was completed, HUD abandoned new city planning and became interested in utility systems for privately developed HUD-financed new communities. These new communities were usually viewed as consisting of a variety of housing types and one or more shopping centers. Establishment of these community centers, however, was impeded by the inability of nearby large cities to supply sewage treatment services and the inability of large power plants to furnish the added electricity needed.

The proposed, community-size, HUD utility system was given the title Modular Integrated Utility System (MIUS). Each MIUS was to be made up of three modules: an electrical generation module, with the heat from the diesel engine or the gas turbine exhaust being used for space heating; a module that burned trash, with useful recovery of heat; and a module for sewage processing. Thus, there would be no dependence on outside sources for other than air, water, and fuel. The initial concept of factory-built, transportable modules gave way to the conventional practice of using systems built on site and sized to meet the needs of the particular development to be served.

Early in the MIUS development, the National Bureau of Standards and NASA became participants in the MIUS Program. They were followed by the AEC, Environmental Protection Agency, and National Academy of Engineering, all of whom also became major participants in the MIUS Program. Interests were broadened to include utilities for military bases, hospitals, and other applications.

When the expectations of oil and gas shortages became pervasive, A. P. Fraas and coworkers began to examine the use of coal in a MIUS system. The coal was to be burned in a fluidized bed, and gas, which would be heated by passing through a heat exchanger in the bed, would be used to drive a turbine for power production. Funding for this concept was provided by both HUD and the Department of Interior Office of Coal Research.

When the Energy Division was formed in 1974 with S. E. Beall as Director, the MIUS work and those engaged in it were transferred to this new division. Persons involved in the program in addition to those mentioned above were E. C. Hise, W. J. Boegly, R. S. Holcomb, J. O. Kolb, M. E. Lackey, W. R. Mixon, G. Samuels, C. L. Segaser, J. V. Wilson, and coworkers.

## 4.11 TECHNOLOGY DEVELOPMENT

Up to this point, Reactor Division history during the 1961 to 1975 period has been discussed primarily in terms of specific quasi-autonomous projects and programs. In many cases, the technology developed in addressing projects and programs had potential for broader application. Examples are (1) fuel rod simulator design and fabrication, (2) equipment and capabilities for in-reactor experiments, and (3) technology and procedures for design and operation of loops for circulating liquid metals and high-melting-point molten salts as well as procedures for safe handling and purity retention of these substances.

In the Engineering Science Section under R. N. Lyon, engineering technology development through discipline-oriented engineering science studies was emphasized. This emphasis provided goals differing from usual project or program goals, that is, providing a system, component, or nuclear complex or addressing a specific situation. At the same time, personnel from this section participated in many of the projects discussed.

The section was divided into two groups, an Applied Solid Mechanics Group under W. L. Greenstreet and a Heat Transfer-Fluid Dynamics Group under H. W. Hoffman. These groups are discussed separately below.

### 4.11.1 Applied Solid Mechanics

The Applied Solid Mechanics Group was formed in the ANP Division in 1954. The purpose was to ensure that the ART components and systems would perform as required under the severe loading and operating conditions to be imposed.

Applied solid mechanics work carried out under both the ANP and GCR programs imbued strong expertise in the applied solid mechanics area. Participation in reactor core and pressure envelope design and analysis on the EGCR Project led to AEC recognition and support in the areas of material behavior and characterization, mathematical modeling of material response to applied loadings, and structural analysis method development. These three elements are key ingredients in structural design technology for designing and assessing adequacy of components and systems.

In the early 1960s, the American Society of Mechanical Engineers was extending its structural design code, the *ASME Boiler and Pressure Vessel Code*,* to cover nuclear reactor systems. The AEC, in cooperation with industry, was participating in support of this extension through the Pressure Vessel Research Committee (PVRC) of the Welding Research Council (WRC), an industry and government agency group that acts as an umbrella organization where mutual problems can be discussed and addressed on a cooperative basis. In this case, development of a nuclear section of the ASME Code (namely, *ASME Boiler and Pressure Vessel Code*, Sect. III, "Nuclear Components") was the common goal.

Pressure boundary components of nuclear power plant systems, particularly those systems essential to safe operation, must be designed so that they will not fail under stresses caused by vibration, fatigue, or demands of pressure, thermal expansion, and mechanical loads. Hence, mainstays of AEC-sponsored work at ORNL in support of ASME Code development included generic-type experimental stress analyses of pressure boundary components and mathematical modeling of responses to loadings to be experienced. This work was initially addressed to responses of pressure vessels subjected to normal system operating conditions.

The work was done both by Reactor Division personnel and through subcontracts with universities and other organizations. Reactor Division participants included R. C. Gwaltney, J. E. Smith, S. E. Bolt, J. W. Bryson, J. P. Callahan, J. M. Corum, S. E. Moore, F. J. Witt, and G. T. Yahr. R. C.

*The *ASME Boiler and Pressure Vessel Code* is a standard that sets minimum safety requirements for the building of components such as pressure vessels, piping systems, pumps, and valves. It has the force of law in the individual states of the United States and is used on an international basis. Nuclear reactor plant designs must comply with this code.

Gwaltney, and J. P. Callahan, in succession, led the work on pressure vessels under a program entitled Experimental and Analytical Investigations of Nozzles Program; this program was initiated in 1963.

A similar program to address piping and piping products was established by AEC in 1967 to support efforts to develop an *American National Standards Institute (ANSI) B31.7 Code for Nuclear Power Piping*. This program was also to be done in cooperation with the PVRC.

S. E. Moore was manager of the resultant Piping, Pumps, and Valves Program, or simply, the Piping Program. The title reflects the fact that AEC assigned the added responsibility of overview of industry-sponsored work on pumps and valves to ORNL. Again, the work was carried out by Reactor Division personnel and through subcontracts. Reactor Division personnel in addition to S. E. Moore included S. E. Bolt, J. E. Smith, W. G. Dodge, R. C. Gwaltney, and J. N. Robinson.

In 1968 and again in 1969 draft versions of the *ANSI B31.7 Code for Nuclear Power Piping* were issued for trial use. ANSI B31.7 became official on August 24, 1969. The piping system design analysis methodology set forth in that code incorporated the earliest results of the ORNL Piping, Pumps, and Valves Program. As additional information was obtained, it was also incorporated. In 1973, ANSI B31.7 was incorporated into *ASME Code*, Sect. III, "Nuclear Components."

Both of these programs were continued under AEC sponsorship until 1974 when the NRC was formed. At that time, AEC safety-related programs were transferred to NRC; the programs on pressure vessels and on piping, pumps, and valves were included. The two programs were merged into one, the Nozzles and Piping Program in 1975 and continued with S. E. Moore as manager.

In 1967, AEC became sufficiently concerned with the safety of reactor pressure vessels used in PWRs to begin shaping a program to further address this area. These steel pressure vessels have walls nearly a foot thick, contain water at an operating pressure of more than 2000 psi, and are subjected to embrittlement by nuclear irradiation. It is crucial that such vessels resist cracking. This concern led to the establishment of the HSST Program to deal with the structural integrity question for LWRs. The program was initiated in 1968.

The HSST Program was also defined with the aid of PVRC and was to be done in cooperation with this organization, which would also provide review and guidance. The program was included under the ORNL Nuclear Safety Program and was under the purview of G. D. Whitman, who was head of the Reactor Division Pressure Vessel Technology Office. F. J. Witt was the first director of the HSST Program and was assisted by J. G. Merkle. The HSST Program is discussed in greater detail in Sect. 4.12.

As stated in the GCRP description, PCRV development work was initiated in 1965 with support from members of the Applied Solid Mechanics Group and others. Participants from the Applied Solid Mechanics Group included J. M. Corum who helped define the program to be pursued, J. E. Smith, W. J. McAfee, and others.

PCRV technology development was pursued when it became apparent that state-of-the-art steel fabrication could not produce vessels of suitable size for the large volume of gas used as the primary coolant in GCR systems. This PCRV Project also is discussed in more detail in Sect. 4.12.

Inelastic behavior at elevated temperatures, such as creep, time-dependent fatigue, and buckling of components and structures, was addressed in connection with both ANP and EGCR work. This experience added significantly to staff expertise.

Graphite mechanics was another area in which expertise was developed under the EGCR Project. Characterization of material response to various stimuli (applied forces, thermal loadings, and irradiation-induced changes) and mathematical modeling of responses were addressed. The characterization was done through testing by both ORNL and Hanford.

In addition to work related to the EGCR Project, a Graphite Mechanics Program was funded by the AEC SNPO in support of a nuclear rocket development program at Los Alamos and, later, a companion program being carried out by the Astronuclear Division of Westinghouse Electric Corporation. The related nuclear rocket work also embraced material behavior characterization and mathematical modeling of response.

Both load-induced deformation behavior and fracture characteristics were studied. Stress and strain data were obtained at room temperature and at temperatures to 5500°F. Fracture behavior investigations spanned the temperature range from room temperature to 4000°F. Both commercial-grade and specialty graphites were examined; the latter required the use of miniature specimens, with diameters down to 1/16 in.

Mathematical modeling of structural responses to applied loadings, which required theory extension, and derivation of fracture prediction correlations applicable to expected loadings were successfully achieved. This program was terminated in 1972 when support for work on the nuclear-powered rocket began to wane.

Graphite Mechanics Program participants included G. T. Yahr, R. S. Valochovic, J. E. Smith, J. G. Merkle, S. E. Moore, R. P. Wichner, J. M. Corum, R. W. Derby, F. J. Witt, and others.

In 1969, the Applied Solid Mechanics Group began work on the LMFBR Program that involved experimental stress analyses of pressure boundary components for liquid metal systems. A need that was not addressed by the LMFBR Program at that time was the development of elevated-temperature structural design technology and a standard for assessing design adequacy; that is, a document providing criteria to determine the adequacy of a component or system for given elevated-temperature service. This was true despite the fact that the FFTF was being designed and built at Hanford, and the Clinch River Breeder Reactor, to be built on the Oak Ridge Reservation, was in the initial design stage. Both had high-temperature liquid metal systems in their design.

Therefore AEC inaugurated work on the development and deployment of structural design technology for LMFBRs in 1970. The purpose was to provide appropriate structural design technology for use throughout the LMFBR Program. The lead role was assigned to the ORNL Applied Solid Mechanics Group, with W. L. Greenstreet to head the ORNL portion as well as the national program on structural design technology development and deployment.

The Applied Solid Mechanics Group was selected because, in the words of AEC, "ORNL had the strongest technical team in the country for this extremely difficult work." At the request of AEC, PVRC established a subcommittee to coordinate ORNL work with that of industry.

Reactor Division participants in the LMFBR Program included J. M. Corum, C. E. Pugh, J. J. Blass, J. A. Clinard, A. G. Grindell, K. C. Liu, W. J. McAfee, R. E. MacPherson, D. N. Robinson, M. Richardson, W. K. Sartory, G. T. Yahr, and H. C. Young. H. C. McCurdy became LMFBR Elevated-Temperature Structural Design Technology Program Coordinator in 1972. R. W. Swindeman and others from the Metals and Ceramics Division ably provided needed support.

The Elevated-Temperature Structural Design Technology Program embraced the following areas: (1) materials properties and behaviors, (2) mathematical analogs for description of material response, (3) structural analysis methods, (4) design acceptance criteria, and (5) confirmatory structural testing. Elevated-temperature design technology for LMFBR systems was initially introduced in full, including design criteria, in 1974 and established precedence by promoting the use of detailed models of inelastic mechanical behavior of materials (constitutive equations) in quasi-routine structural design analyses. It not only provided a rational basis for elevated-temperature structural design in the United States, it provided a basis for international elevated-temperature structural design code development. Work on extensions and improvements have continued under the ORNL program since that time.

### 4.11.2 Heat Transfer–Fluid Mechanics Group

The Heat Transfer–Fluid Mechanics Group was the descendant of the Heat Transfer and Hydrodynamics Section that was formed in the Reactor Experimental Engineering Division in 1952 with H. F. Poppendiek as leader; this section was to support nuclear reactor development activities at ORNL.

When the Heat Transfer and Hydrodynamics Section was formed, ORNL attention was focused on homogeneous reactor concepts and on the higher-temperature ARE and ART aircraft power reactor systems. The problems requiring attention involved the generation, transfer, transport, and rejection of heat at higher temperatures and heat fluxes (or flow rates) than had generally been confronted to that date. Heat generation within the fluid added another new facet. Along with this, there was the need for fundamental data on thermal properties of new classes of heat carrier fluids, namely molten salts and liquid metals, that could withstand the more stringent conditions required by nuclear power generation. These efforts at ORNL led the way for the country.

Undergirding specific needs of present and future reactor projects was the long-term research goal in thermal science technology. Advantage was taken of each opportunity and project to move heat exchange technology forward by providing design data, developing innovative measuring techniques, and by supporting development of basic understanding of thermal and flow phenomena. In the process, as national goals shifted, the directions changed from concentration on homogeneous nuclear reactors to other reactor concepts, to reactor thermal safety, to desalination, to space power, to thermal pollution, and to energy conservation. The descriptions that follow highlight some of the contributions to thermal engineering science.

#### 4.11.2.1 Reactor Thermal Technology

The group supported the MSR development project through generation of heat transfer and thermal property data specific to the MSR design and in data interpretation from system component tests. Among those involved were H. W. Hoffman and S. Cohen for heat transfer and W. Powers and S. I. Kaplan for thermophysical properties.

A fundamental finding was the effect of interfacial (coolant-metal wall) thermal cycling in molten salt systems on the integrity of the containment wall itself. These fluctuations, which are embedded in the nature of turbulent flow and the flow channel geometry, caused either chromium removal along grain boundaries of the container material and extensive, deep wall cracking or deep cracking alone, as determined by seminal experiments by H. W. Hoffman and J. J. Keyes, respectively. J. J. Keyes discovered the important fact that the product of the temperature oscillation amplitude and the frequency of oscillation was a constant, whose value could be used to predict the service lifetime of a container.

In the case of the EGCR, studies (led by J. L. Wantland, W. J. Stelzman, and G. J. Kidd, Jr.) were conducted on fuel assemblies, which were bundles of metal-clad fuel rods mounted in graphite sleeves and stacked vertically in the reactor core. Objectives were to determine the effect of midbundle rod spacers on the heat extraction from each fuel array and the impact of rod bundle rotation between successive sleeves stacked within a reactor core channel. This led to the development of an experimental technique for determining heat transfer coefficients (measures of heat flow propensity) using naphthalene (a crystalline hydrocarbon) and to generalized mapping of gas velocities within and exterior to rod bundles of limited array and various spacing.

In the case of the MPRE, the thermal technology demands were severe and led to the identification and characterization of new phenomena and the development of new system components. Because of the high chemical reactivity of the potassium coolant and, hence, its extreme wettability (covering of a surface) on confining metal surfaces, the energy required to initiate boiling (superheat) is about ten times greater than for water. Release of thermal energy associated with the superheat on sudden boiling inception could cause severe pressure transients and, hence, damage of the system including breach. Boiling initiation was quantified, and means for lowering the required superheat levels were studied. These pioneering studies were conducted by A. I. Krakoviak, H. W. Hoffman, and J. A. Edwards. In addition, important studies were done on liquid and vapor separation to identify equipment geometries applicable to near-zero gravity circumstances of earth orbit space (J. J. Keyes).

An outgrowth of the thermal cycling and incipient boiling studies was development and testing of hot-film probes for precise measurement of temperature fluctuations in fluids flowing along very smooth surfaces. This contributed to the understanding of both interfacial fluid-dynamic phenomena and the effects of surface wetting. Studies in this area were done by J. J. Keyes and R. P. Wichner.

Another interesting project dealt with the experimental characterization of flow dynamics of a vortex-plasma, fission reactor proposed for advanced power applications. This very high temperature system was to produce thrust through the expulsion of high-temperature hydrogen. To achieve the required heating, a gaseous uranium compound was injected into a strongly rotating hydrogen flow. The rotation kept the uranium within a nuclear-energy-wise critical region; the flow of the hydrogen stream kept the hot plasma away from the containing wall and provided for highly efficient direct-contact heat removal from the fissioning region.

In the laboratory, the reactor was simulated with cold helium and argon flows by J. J. Keyes and R. E. Dial. The data obtained supported sophisticated analytic and computational techniques derived by T. S. Chang and W. K. Sartory, for describing and predicting the thermal and flow behavior of the reactor. Studies by G. J. Kidd, Jr. on uranium loss from such a system used new flow visualization techniques in full-scale models and added greatly to basic knowledge of thermal and bypass flow behavior.

A significant amount of work was done in connection with the ORNL PWR Blowdown Heat Transfer Separate Effects Program. Important contributions were made by H. W. Hoffman, D. G. Thomas, W. G. Craddick, R. E. Bohanan, J. D. White, and R. A. Hedrick to the thermal design of the fuel rod simulators for modeling nuclear fuel rod performance; to means for measuring internal rod temperatures and, from these, extracting surface temperatures; and to computational techniques for predicting rod transient flow and pressure distributions radially and axially in full-scale sections of simulated PWR fuel-rod arrays. These results were extremely important in developing the data base obtained on high-pressure, high-temperature, transient phenomena associated with reactor coolant blowdown.

#### 4.11.2.2 Desalination

Work under the Nuclear Desalination Program was undertaken largely as a vehicle for supporting an underlying goal of developing techniques and devices for enhanced heat transfer, while satisfying material purposes. Outstanding contributions were made in the areas of distillation, led by L. G. Alexander and H. W. Hoffman, and membrane separation processes for water desalination, led by D. G. Thomas. In the case of distillation, the contributions were those of defining, constructing, and testing tubes with special (convoluted) surfaces to optimize evaporative performance; for membranes, the contribution, principally by J. D. Sheppard and J. S. Watson, included identifying and refining techniques for forming and repairing membranes on cheap tubular substrates. During this same period, basic work was done by D. G. Thomas in observing swirling or vortex flow interactions downstream of obstructions and in controlling interaction effects at the bounding surface.

#### 4.11.2.3 Geothermal and Ocean Thermal Energy Conversion

The desalination experience led directly into studies for improving thermal performance of alternate energy systems based on geothermal heat* and ocean vertical temperature difference† sources. Enhanced heat transfer surface geometries developed in the desalination studies were extended effectively for extracting useful heat from these two heat sources.

Experimental studies, which were begun in the mid-1970s, were continued throughout the remainder of the decade. They were led by R. W. Murphy. Analyses and evaluations by R. N. Lyon and S. L. Milora contributed to the effort. Knowledge developed regarding working fluids (fluorocarbons and low-boiling hydrocarbons for geothermal systems and ammonia for ocean thermal systems) added substantially to the thermal technology data base on enhanced heat transfer processes.

#### 4.11.2.4 Thermal Pollution

The Nuclear Desalination Program led to the consideration of so-called "power parks," comprising five of more 1000-MW(e) nuclear plants that would supply electricity over a very large surrounding region. Within this context, studies were undertaken to determine the impact on local and regional weather of the large thermal releases to the atmosphere from the stacks and cooling towers of a park. H.W. Hoffman led in the origination of these studies in the mid-1970s; they were carried to successful completion under the direction of A. A. N. Patrinos with assistance from N. C. J. Chen, L. Jung, and others.

A major field study was done with the cooperation of the Georgia Power Company at its Bowen power station near Cartersville. This was then the largest coal-fired plant [four units with a total output of 3160 MW(e)] in the United States. The equipment and techniques developed for this study were later applied to a broader national acid-rain program.

#### 4.11.2.5 Energy Conservation

The 1961 to 1975 period ended with the Heat Transfer–Fluid Mechanics organization verging on major contributions to the utilization of waste heat from utility and manufacturing processes and from natural solar heat and environmental cold sources. There was a flowering of ideas nationwide under the impetus of the global oil crisis. Therefore, the organization's work became that of managing a national effort on thermal energy storage by heat absorption associated with phase changes of materials (particularly freeze-melt‡). H. W. Hoffman was the first leader of this still continuing effort. Important applications were to be toward (1) load-leveling for electric utilities and (2) the collection, retention, and transfer of heat from its time of generation to its time of use on a daily or seasonal schedule.

### 4.11.3 Epilogue

In 1971, G. D. Whitman replaced R. N. Lyon as section head, and the Heat Transfer–Fluid Mechanics Group under H.W. Hoffman became the Heat Transfer–Fluid Dynamics Section. W. L. Greenstreet was replaced as group leader by J. M. Corum in 1974.

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*Available heat from the earth's interior.

†Due to natural solar heat collection.

‡For example, the heat absorption associated with ice to water transition, a phase change.

## 4.12 PRESSURE VESSELS

Two programs are described under this heading. They are the PCRV Program and the HSST Program.

### 4.12.1 PCRV Program

Concrete pressure vessels for nuclear plants provide an advantage because they can be built in place, precluding the need for transporting large vessels to the sites where they will be used. This makes them particularly attractive for large GCR use, which requires very large pressure vessels.

A concrete pressure vessel for this purpose, unlike a steel pressure vessel, is an assembly of many components and, therefore, a complex structure. It is generally made up of reinforcing steel in the form of rods and prestressing tendons as well as concrete and has a metal liner to maintain internal pressure and prevent escape of the contained medium. This liner also transmits pressure-induced forces to the prestressed concrete structure. The reinforcing rods are embedded in the concrete, while the tendons are installed in passages in the vessel wall and can be tensioned in place to prestress the structure.

Because of the structural complexity, an adequate capability for analysis of the strength and structural behavior, including time-dependent deformation and fracture, is necessary, as in the case of other components of a nuclear reactor system. Thus the PCRV Program was established at ORNL in 1966 to examine the characteristics of concrete and the problems that these characteristics might present from a design and operation standpoint, to begin using modeling techniques in the design of vessels, and to establish analysis methods by which the vessels can be designed with confidence.

The work in these areas included development and improvement of computer programs for analyzing concrete structures, development of high-temperature capabilities for testing concrete, determination of properties of concrete and metallic materials, and evaluation of acoustic emission and other instrumentation techniques for monitoring concrete pressure vessels. Studies of models were conducted to evaluate and demonstrate the usefulness of small models in determining structural behavior of PCRVs when subjected to internal pressures at levels through ultimate loading levels. These models were instrumented to measure deformations at selected locations and the onset of cracking at locations of major interest. Vessel regions that are structurally complex and least amenable to exact analytical treatment were subjected to detailed investigations through studies on models. Thus, in total, the program addressed analytical methods development and validation, materials and materials behavior, determination of vessel model and individual structural feature responses to projected loadings, and examination of instrumentation for operational behavior monitoring.

The single largest task undertaken in the Reactor Division under this program was the design, construction, and testing of a 1/6-scale, single-cavity PCRV model that was subjected to simulated HTGR operating conditions. The model was a thick-walled cylinder with a height of 48 in., an outside diameter of 81 in., and a wall thickness of 18 in. It was prestressed both axially and circumferentially and subjected to an internal pressure of 700 psi and a temperature differential, varying from 150°F on the inner surface to 75°F on the outer surface. The model was held at these simulated normal operating conditions for 460 d. The model then was subjected to an off-design, superposed, hot spot of 450°F for an additional 84 d. The test and associated analyses demonstrated that long-term, time-dependent behavior can be predicted and that basic structural integrity can be maintained under localized heating conditions.

The PCRV Program has made important contributions to concrete pressure vessel technology and has been extended to address applications other than for GCR pressure vessels. It remains an active program.

The studies under this program were also conducted by ORNL and subcontractor personnel. Reactor Division contributors to this program, in addition to G. D. Whitman, who was coordinator, are J. M. Corum, G. C. Robinson, J. E. Smith, W. J. McAfee, J. P. Callahan, W. G. Dodge, D. J. Naus, M. Richardson, and others. D. A. Canonico, J. C. Griess, R. K. Nanstad, W. J. Stelzman, and J. G. Stradley were among the contributors from the Metals and Ceramics Division. Important contributions were also made by Instrumentation and Controls Division personnel.

### 4.12.2 HSST Program

Early in 1967, the HSST Program commenced as an outgrowth of recommendations contained in a letter, dated November 25, 1965, from the Advisory Committee on Reactor Safeguards (ACRS) to AEC. This letter identified the fact that LWR nuclear power plant safety is generally predicated on the avoidance of sudden large-scale rupture of the reactor pressure vessel. The ACRS suggested that the industry and AEC should give further attention to methods of stress analysis, development of inspection methods, and improvements in the means of evaluating the factors that could affect the propagation of flaws during vessel life to enhance the argument for incredibility of vessel failure.

These recommendations produced intensive planning for ~1 year by the industry and government. This activity was led by the PVRC and culminated in the HSST Program, which has continued as a focal point for nuclear pressure vessel integrity R&D since that time. F. J. Witt was Program Director from 1967 to 1973, followed by G. D. Whitman, 1973 to 1982.

The first HSST Program plan, issued on April 1, 1968, gave sufficient detail to provide background information, objectives, logic charts, project descriptions, and schedules organized around 11 tasks:

1. Program Administration and Procurement;
2. Materials Inspection and Control;
3. Characterization of HSST Program Plates for Testing;
4. Characterization and Variability of Plates, Weldments, Forgings, and Other Material and Product Forms;
5. Transition Temperature* Investigations;
6. Fracture Mechanics Investigations;
7. Fatigue and Crack Propagation Investigations;
8. Irradiation Effects;
9. Complex Stress States;
10. Periodic Proof Testing and Warm Prestressing;† and
11. Simulated Service Tests.

The HSST Program covers three broad areas of investigation involving materials, analysis, and experimental validation, with efforts closely coordinated to result in an effective program on solid mechanics R&D. The program, which initially was under the AEC Nuclear Safety Program, was transferred to the NRC in 1975. This program has been highly successful; it has earned broad support from industry and government alike and has been given the opportunity to produce data vital to the safe operation of nuclear power systems.

Additional personnel involved in the program included J. G. Merkle, G. C. Robinson, L. F. Kooistra, J. E. Smith, A. A. Abbatiello, R. H. Bryan, J. H. Butler, R. D. Cheverton, R. W. Derby, P. P. Holz, R. W. McCulloch, and C. L. Segaser. Metals and Ceramics Division personnel including D. A. Canonico, S. K. Iskander, R. K. Nanstad, W. J. Stelzman, and W. R. Corwin played important roles, as did B. R. Bass and others in the Computer and Telecommunications Division.

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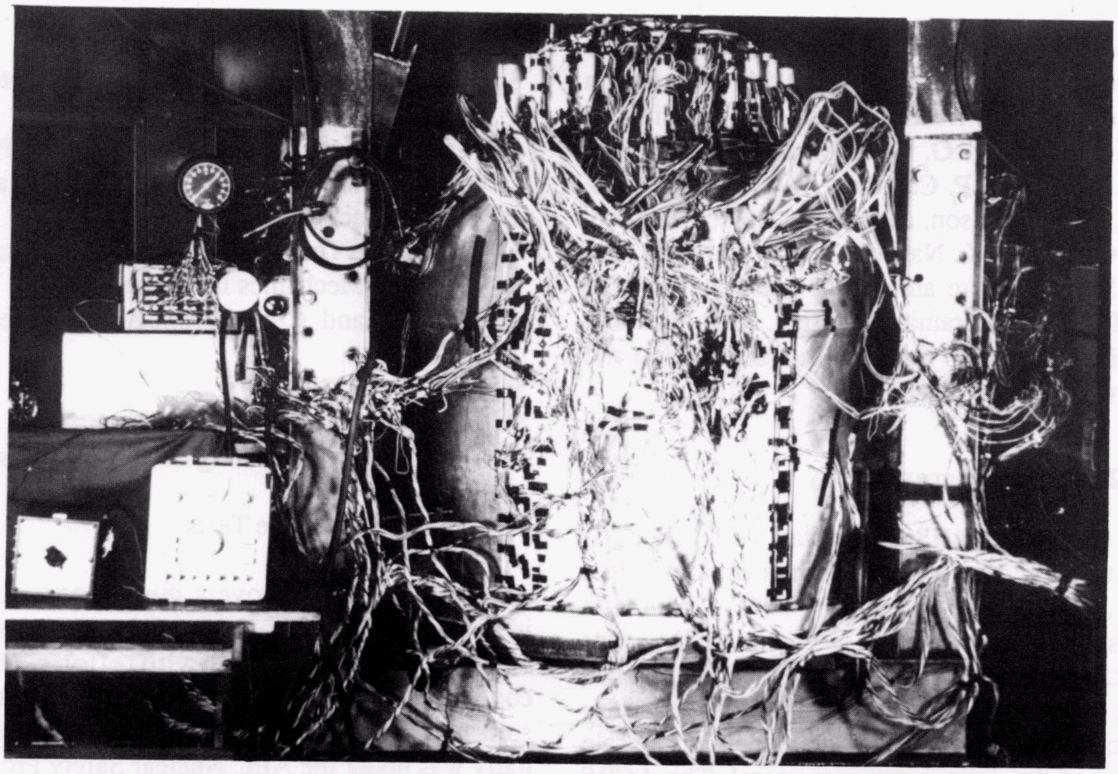
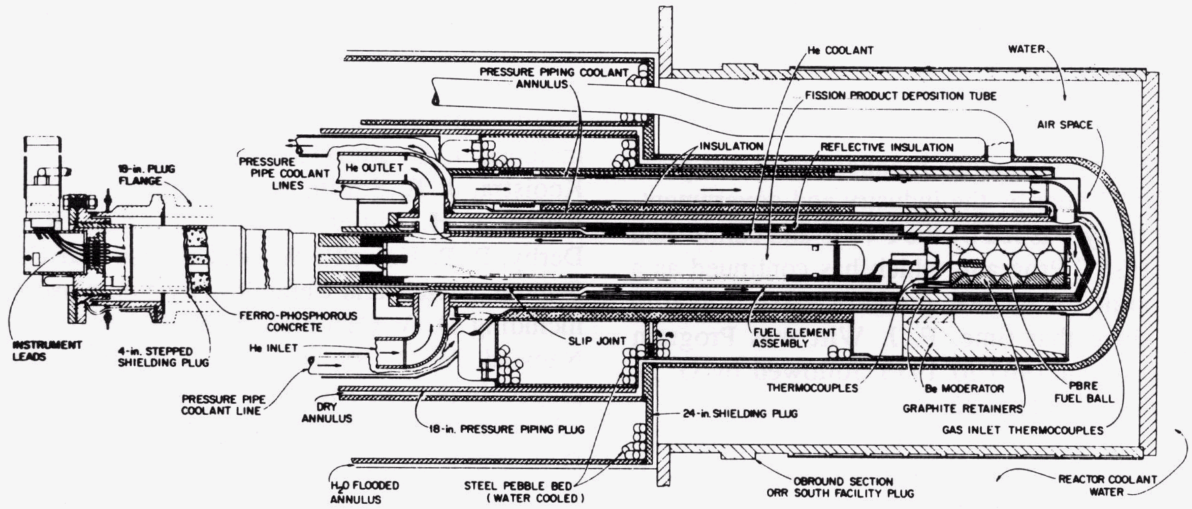
*Temperature at which failure changes from brittle to ductile.

†Warm prestressing behavior is the term commonly used to describe an apparent increase in resistance to fracture of pressure vessel steels resulting from a previous loading at a higher temperature.

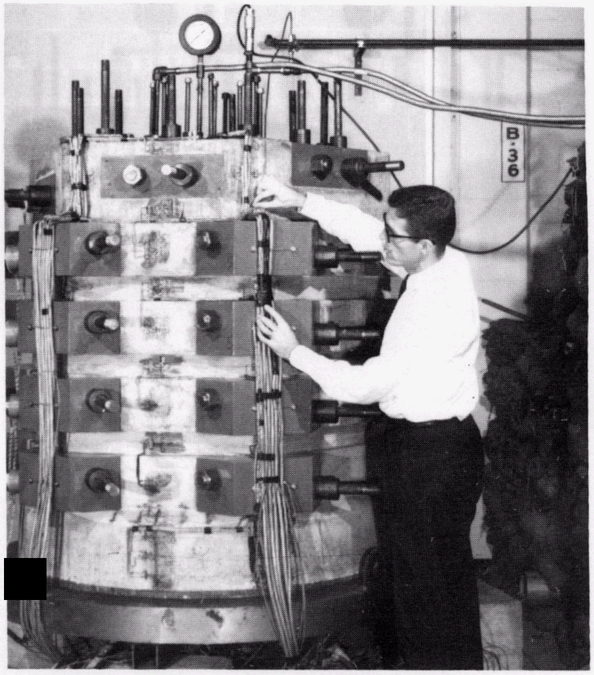


*The pressure vessel for the EGCR was cylindrical with hemispherical heads. The top head, as originally designed, had a cluster of 53 nozzles plus two instrument and two gas outlet nozzles. A 1/5.533 scale model of this structurally complex top head was built, instrumented, and tested. The instrumented vessel is shown mounted in a frame and ready for testing.*

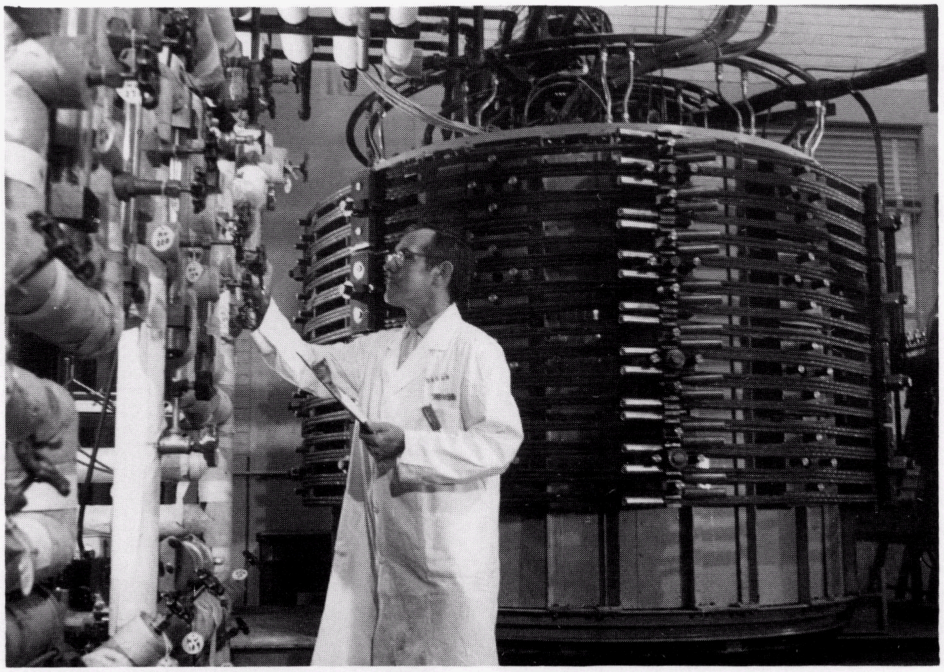



*Irradiation testing of both liquid and solid fuels was begun in the division in the 1950s and persists today. Irradiation testing of materials, such as graphites and stainless steels, was added relatively recently. Testing has been done in the MTR, LITR, Engineering Test Reactor (ETR) (at the National Reactor Testing Station in Idaho), ORR, High-Flux Isotope Reactor (HFIR), High Flux Reactor (HFR) (at Petten, The Netherlands), and the Ford Reactor (at the University of Michigan). Shown are the in-reactor portion of a loop assembly for testing fueled-graphite spheres in the ORR.*

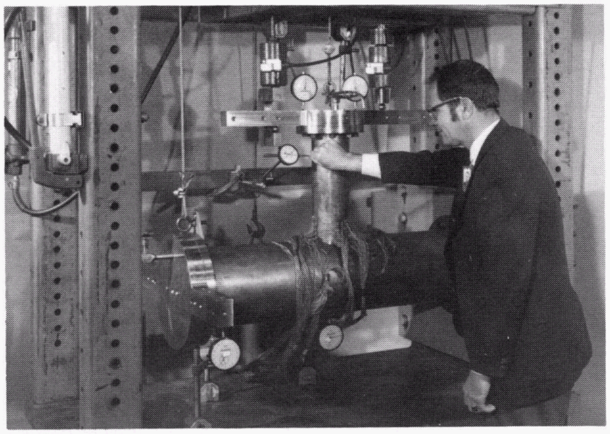
***The Reactor Division's entry into the area of concrete materials and structures was in connection with prestressed concrete reactor vessels for gas-cooled reactors (GCRs) and was guided by a prestigious advisory committee of five national experts, shown meeting here in 1966. In the front row (left to right) are Professor Boris Bresler, University of California, Berkeley; Professor Clyde Kesler, University of Illinois; Arthur R. Anderson, ABAM Engineers, President of American Concrete Institute; Bryant Mather, U.S. Army Corps of Engineers Waterways Experiment Station. In the back row are Grady Whitman; Jim Corum; Eivind Hognestad, Portland Cement Association; Don Trauger; and Mike Bender.***



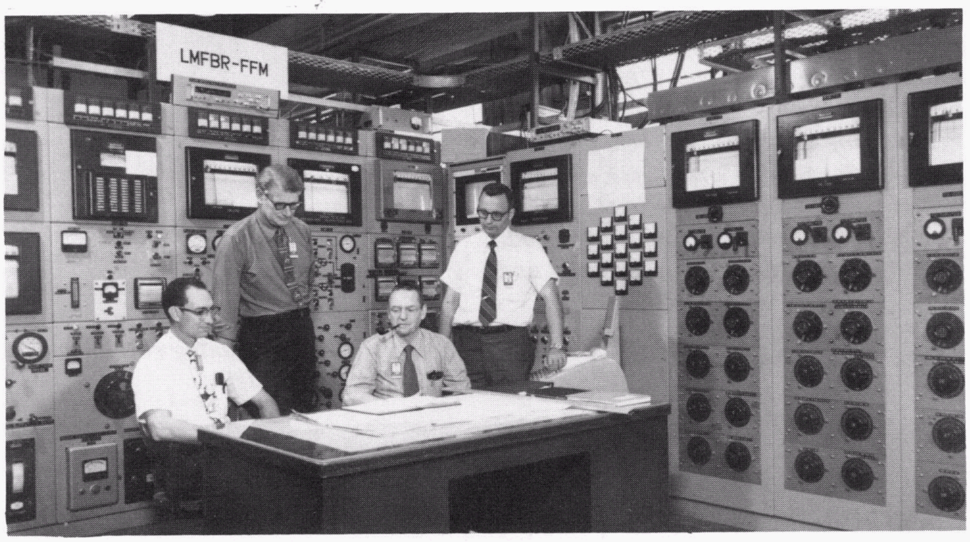
***Studies were conducted to evaluate and demonstrate the usefulness of small models in determining structural behaviors of prestressed-concrete reactor vessels for GCRs under in-service loadings. J. M. Corum is shown making readiness evaluations of an instrumented model.***



***H. D. Curtis in front of 1/6-scale HTGR prestressed-concrete reactor vessel model being fabricated for testing under long-term mechanical and thermal loading.***

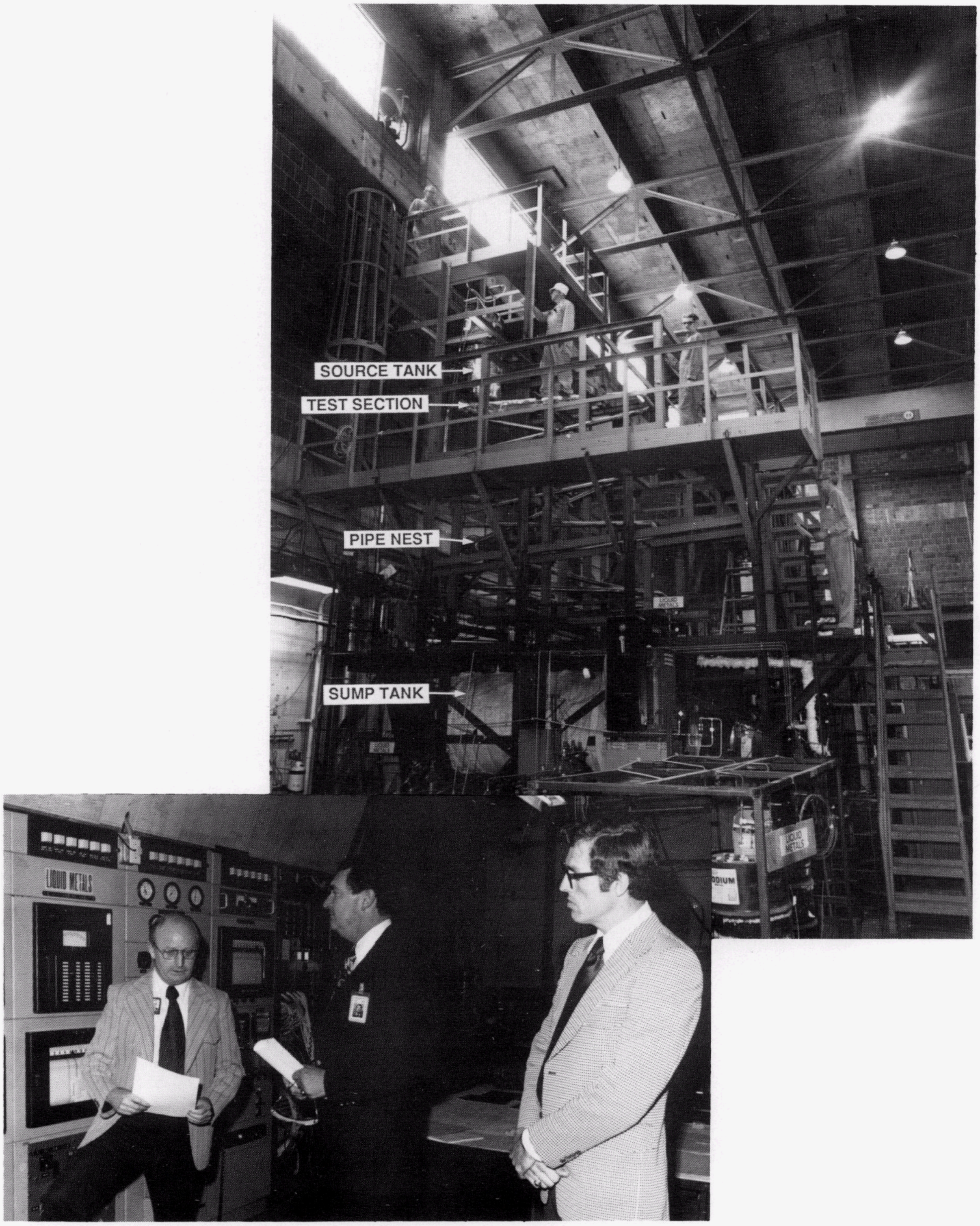


*S. E. Bolt making an adjustment to instrumented nozzle-to-shell attachment for obtaining data to be used in analysis method assessments.*



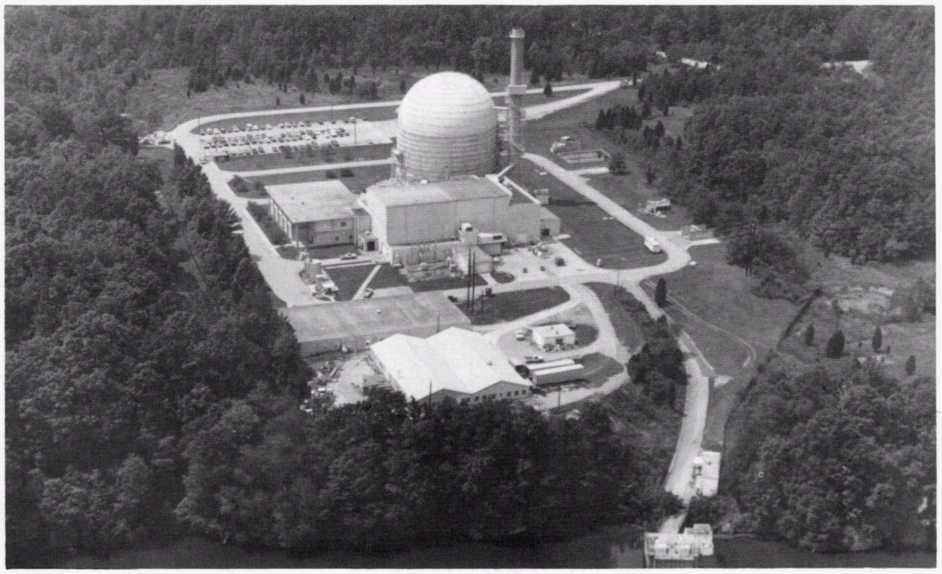

*From left, P. A. Gnadt, R. E. MacPherson, J. L. Wantland, and M. H. Fontana in front of instrument and control panels for the FFM, a high-temperature thermal-hydraulic sodium test loop used to study flow and heat transfer in LMFBR cores.*

*Models of reactor pressure vessels intentionally flawed and tested to failure, such as this one, have demonstrated adequate integrity and safety margins of in-service vessels. J. W. Teague is pictured measuring width of a crack.*

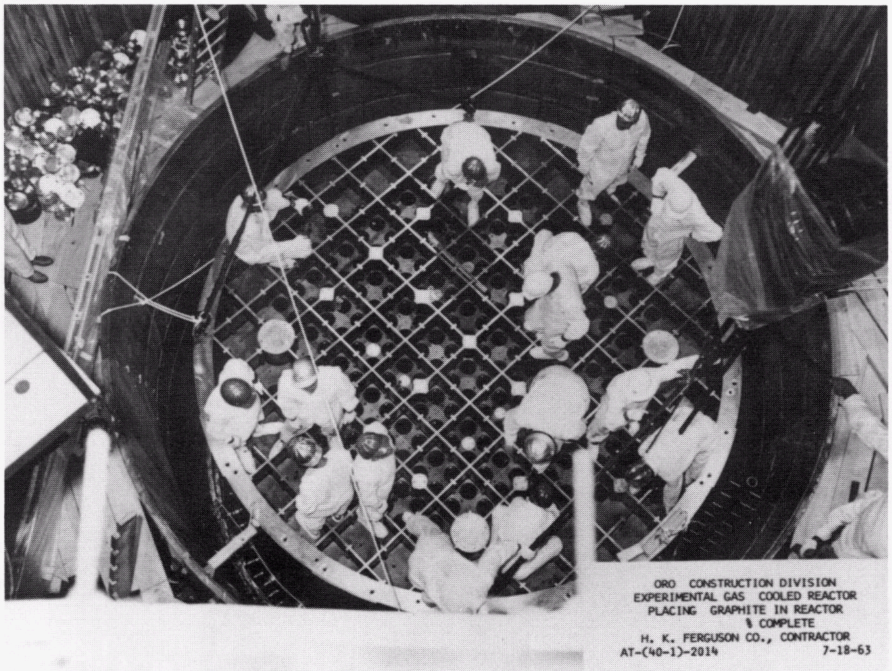



*W. H. Duckworth (on first landing) during construction of the Thermal Transient Test Facility, a sodium loop built and operated as part of the LMFBR High-Temperature Structural Design Program. Inset: W. L. Greenstreet (left) and J. M. Corum (right) briefing T. A. Nemzek (Director of Reactor Research and Development Division, Atomic Energy Commission) at control panel.*

*The Experimental Gas-Cooled Reactor (EGCR), built on the Oak Ridge Reservation, was to demonstrate the power production capability of a gas-cooled nuclear reactor and to obtain information that could be applied to the design and operation of future reactors. It was designed to be operated either solely as a power generation facility or as a combined power-generating and experimental facility. On January 7, 1966, just before fuel loading, the project that was begun in 1959 was terminated by the Atomic Energy Commission.*



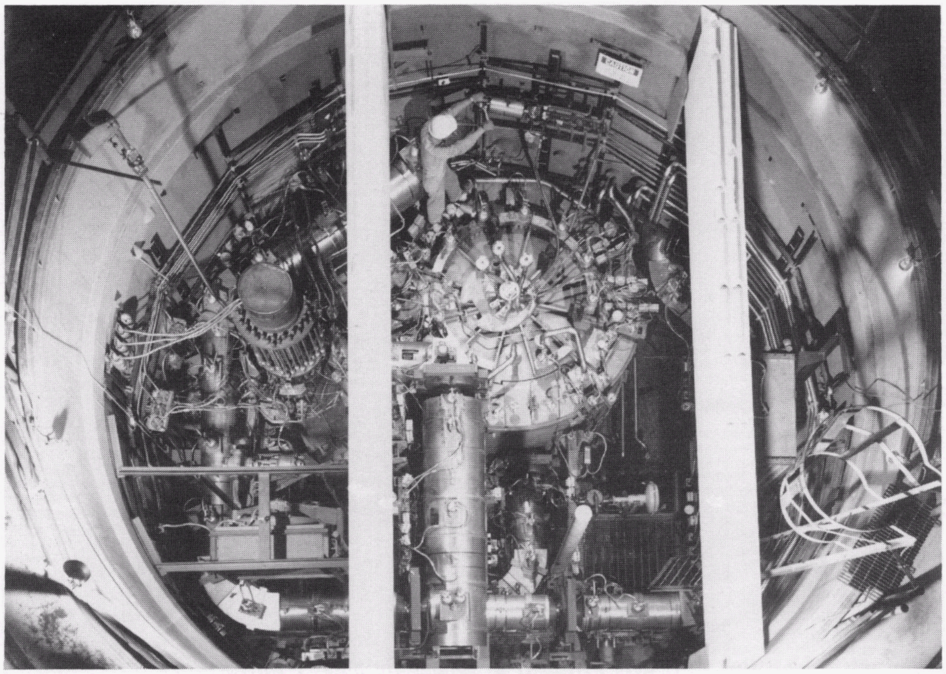
*Aerial view of EGCR complex from above Melton Hill Lake. The office and control building is on the left. The reactor containment building is the circular building with the dome on the right.*




*Workmen installing square columns of graphite in the reactor core. Fuel elements and control rods were to be placed in the holes in the graphite columns. Reactor Division Personnel assisted in ensuring the core's structural integrity. These activities launched graphite structural mechanics work that continues in the Engineering Technology Division today.*

***The Molten-Salt Reactor Experiment (MSRE) was a derivative of the aircraft nuclear propulsion work in connection with the ARE and ART. It was used to demonstrate that the desirable features of the molten-salt concept could be embodied in a practical reactor that could be constructed, operated, and maintained with safety and reliability. The reactor first became critical on June 1, 1965; it was shut down permanently in December 1969 after successful demonstration of operating capabilities.***

***Final calculations and last-minute checks are made at the control panel before achieving initial criticality. Seated around the console are, from left, J. R. Engel, J. E. Wolfe, J. L. Crowley, P. N. Haubenreich, and W. C. Ulrich; standing, from left, are J. Emch (leaning on console), C. D. Martin, Jr., R. L. Moore, G. H. Burger, E. B. K. Ohrenstein (Bunker-Ramo, computer manufacturer), J. Schmith (head down), H. R. Payne, and W. H. Duckworth.***

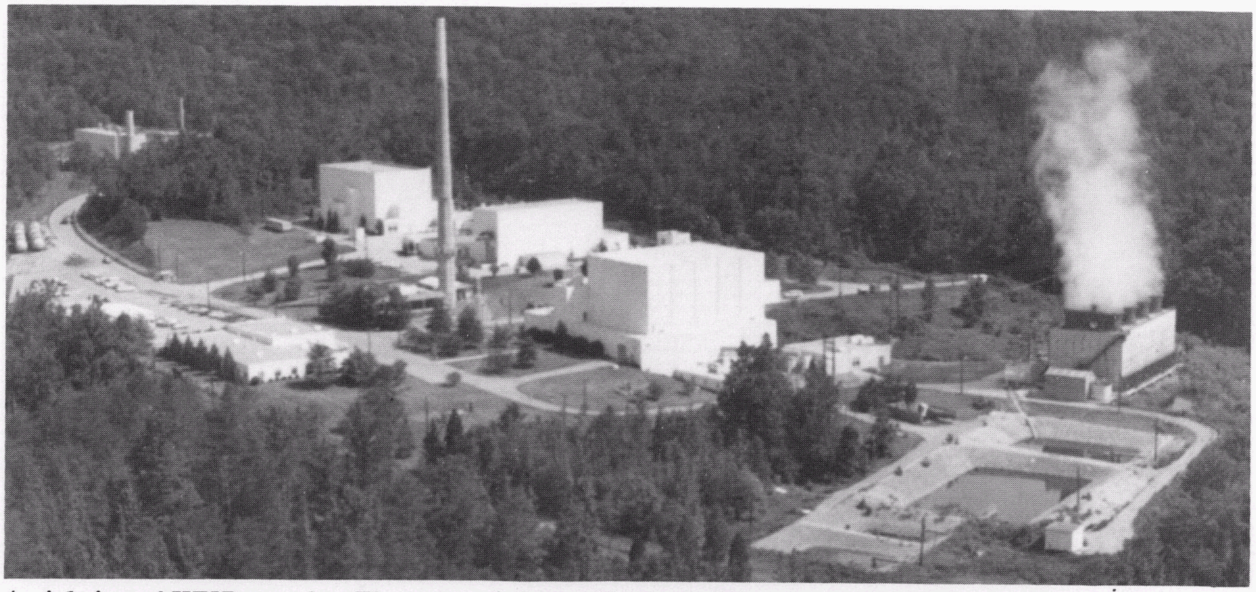


***View looking into the reactor chamber of the MSRE. Major items are the reactor itself, the primary fuel pump, and the primary heat exchanger for removing heat from the fuel.***

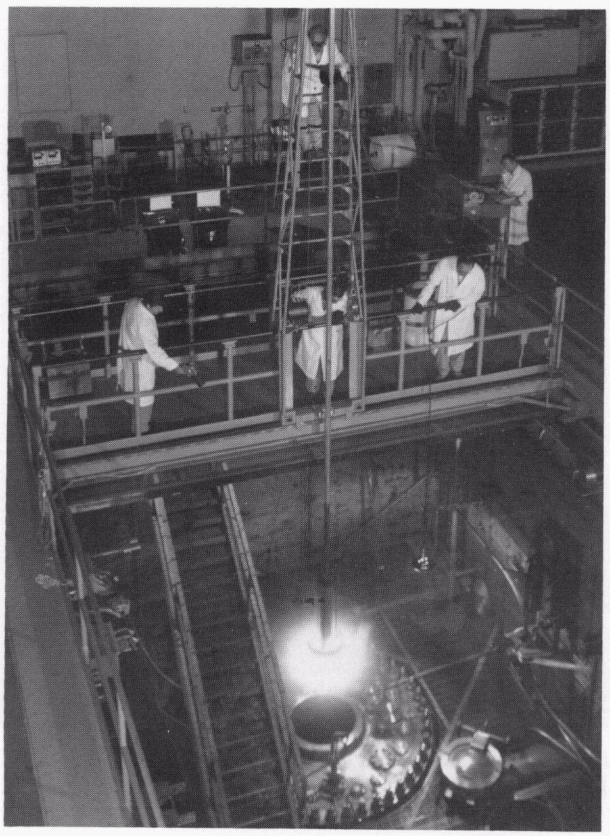
*MSRE shutdown after operating 100 consecutive days for removal and examination of graphite and metal samples placed in the core. Seated at the console, left to right, are R. B. Briggs and E. S. Bettis; standing, left to right, are P. N. Haubenreich, M. Richardson, and J. L. Crowley.*

*The MSRE, fueled with uranium-233, was brought to power by AEC Chairman G. T. Seaborg on October 8, 1968, with R. W. Stoughton, codiscoverer of uranium-233, looking on. Others at the console are, from left, A. I. Krakoviak and J. R. Engel. Standing at the end of the console is A. M. Weinberg.*

*The HFIR was designed primarily to produce chemical elements heavier than uranium (transuranium elements); uranium is the heaviest element that occurs naturally on earth. The HFIR is also extensively used to study irradiation effects on materials and has been an outstanding research tool; it first reached criticality in August 1965 and has operated successfully since that time.*

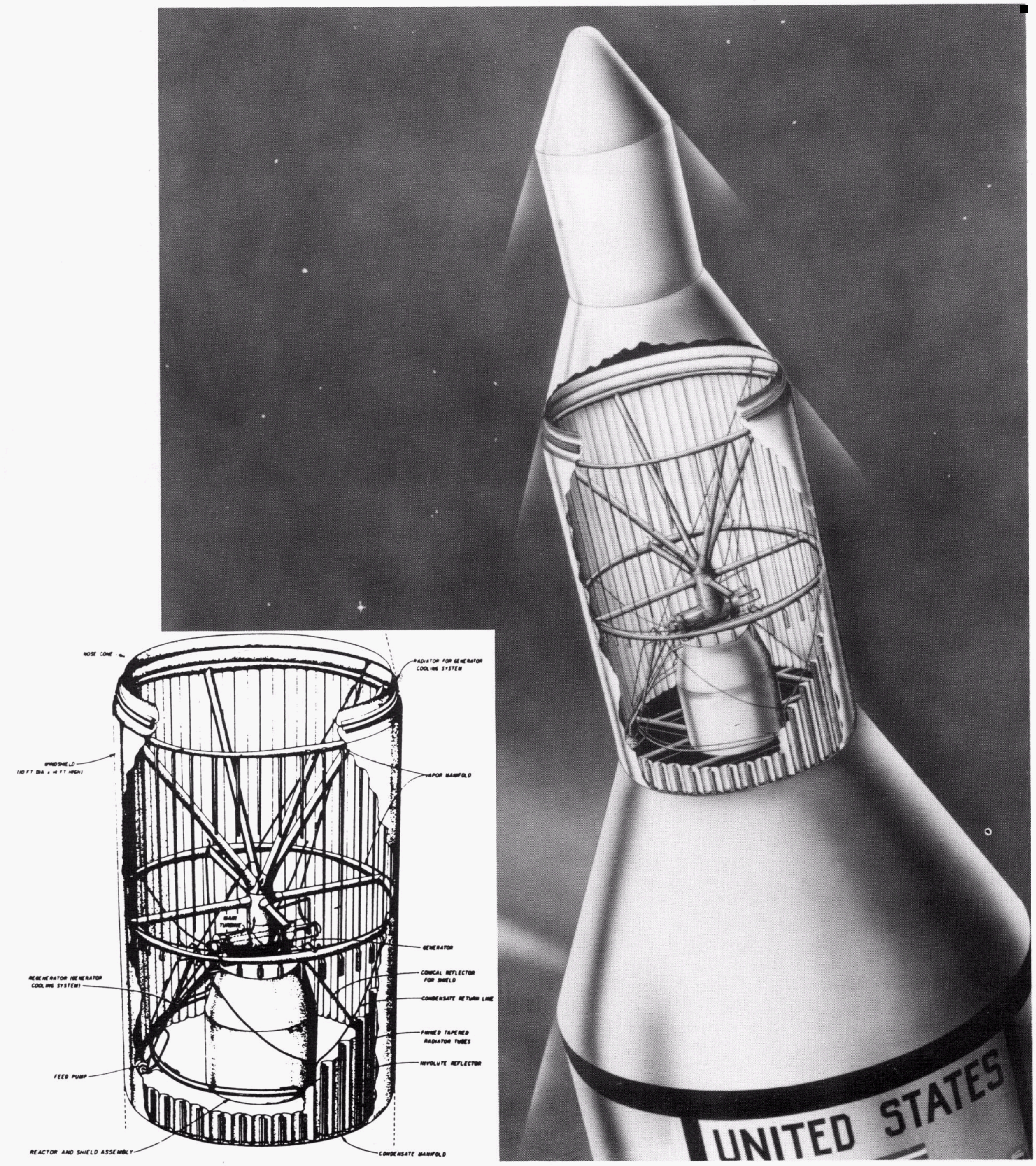


*Aerial view of HFIR complex. The reactor is housed in the building on the right. Reactor heat is dissipated to the atmosphere through use of the cooling tower on the far right.*



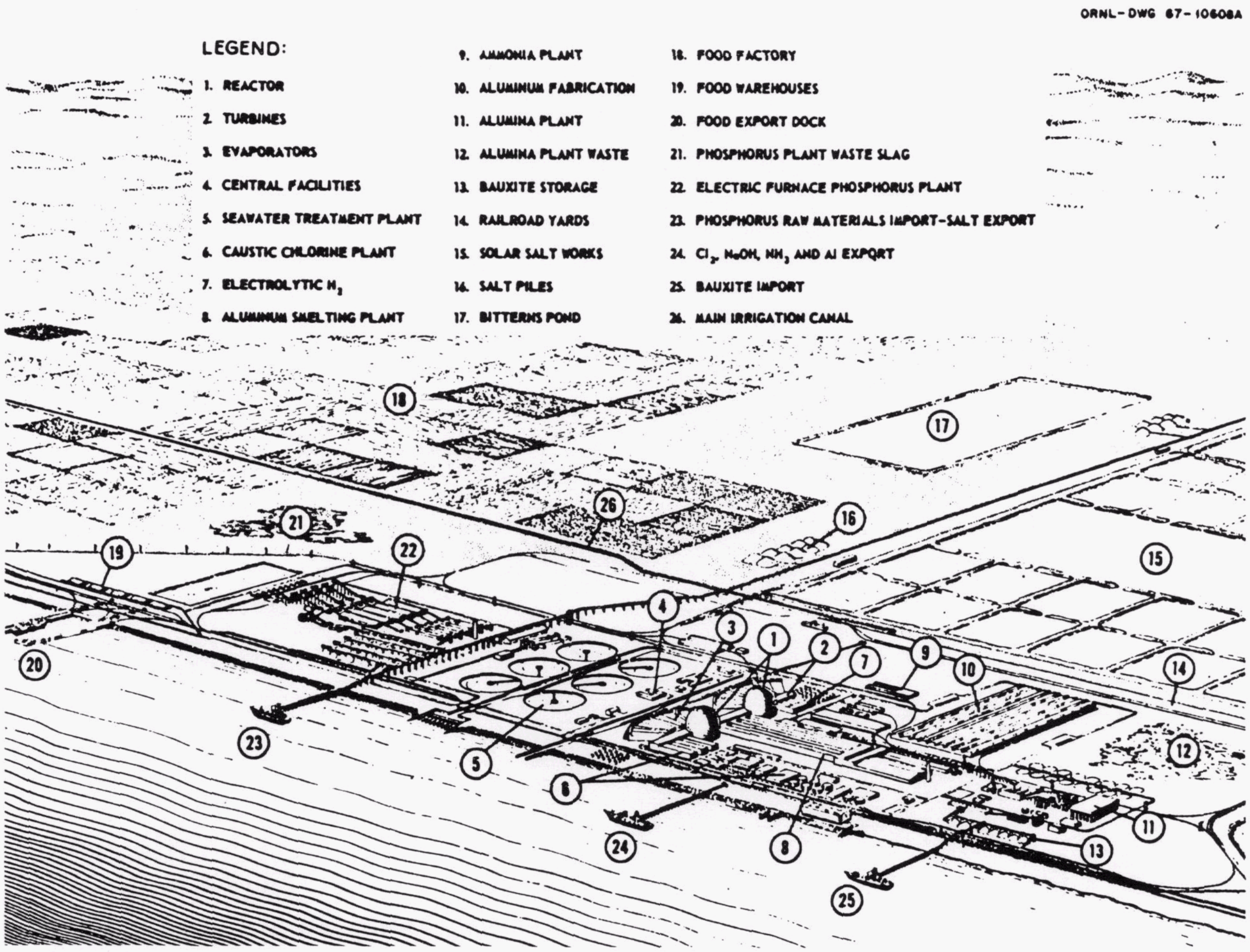
*Fuel element being removed from the reactor. This reactor also emits the blue glow of Cerenkov radiation.*

***The Medium Power Reactor Experiment (MPRE) was planned to be a forerunner of a nuclear power plant for space applications. Design of the full-scale reactor proceeded concurrently with component tests. The overall project was initiated in 1958 and was terminated in 1966.***



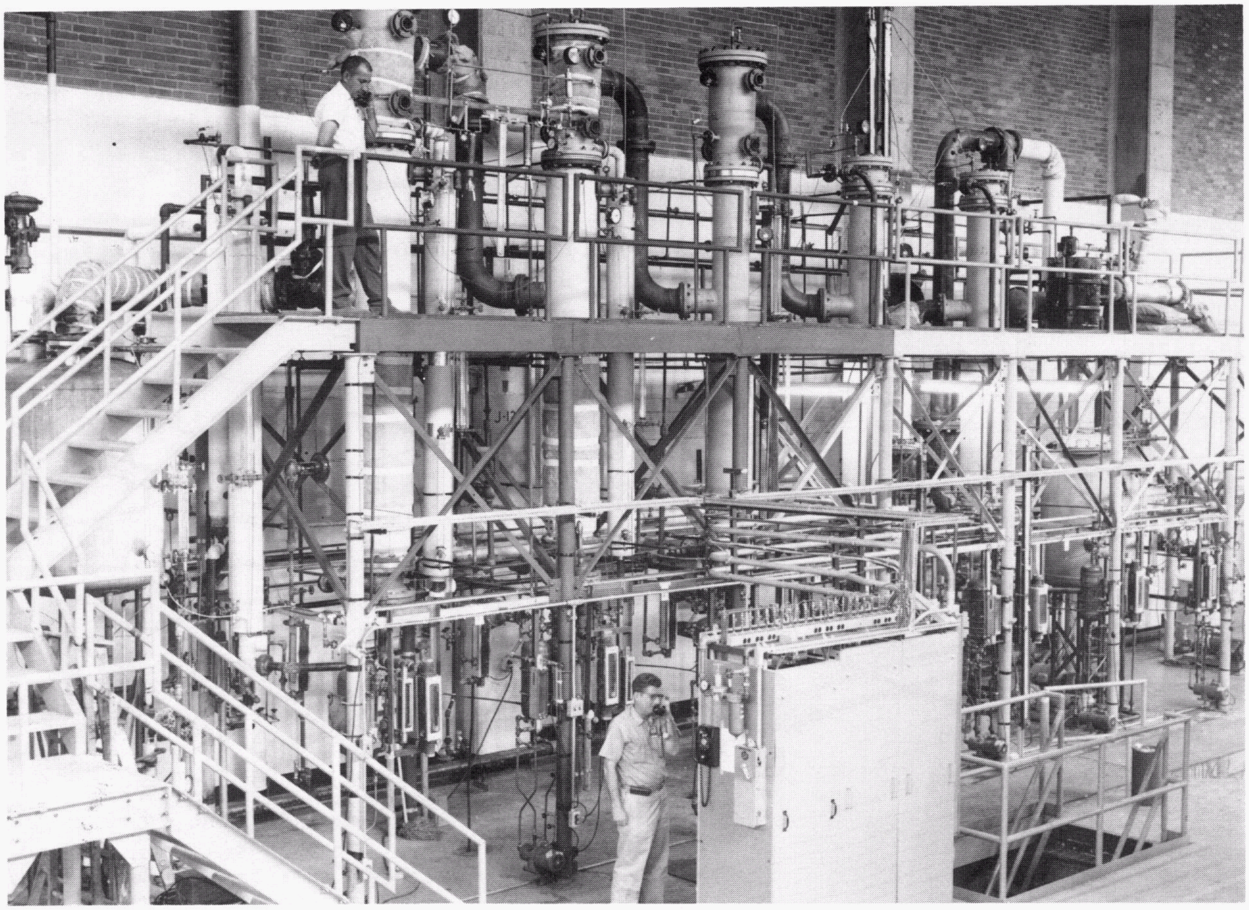

***The MPRE launch package mounted on a Titan II launch vehicle is depicted in this artist's drawing. The launch package includes the space power module surrounded by radiators for heat dissipation and a windshield for launch protection.***

***Between 1965 and the early 1970s, agro-industrial complexes were envisioned to include a nuclear water-desalting complex located in an arid region where the product water could be used for intensive agricultural production and further support could be derived from a companion industrial complex for producing fertilizer and other possible products. Study and development of this idea were pursued to examine the use of such complexes in the Middle East, India, and other places to both support people and enhance the quality of life.***




***A schematic view of an agro-industrial complex is shown along the shoreline of a saltwater source. This complex desalts water for irrigation of crops and other uses; salt is reclaimed and exported. It produces fertilizer ingredients for food factory use and ammonia, sodium hydroxide, chlorine, and aluminum for sale. Aluminum product fabrication is also carried out.***

*Vertical tube evaporators (VTEs) for water desalination were studied extensively in connection with desalination studies in general. The objective was to improve VTE performance.*



***Shown is a multistage VTE that was built and operated in Building 9204-1 as a part of the Desalination Program that was pursued, beginning in the mid-1960s. G. Windle is on the upper platform, and J. Hurst is on the floor below.***

# 5. 1975–1992 DEVELOPMENT AND APPLICATION OF ENGINEERING TECHNOLOGY

The period from 1975 to 1992 presented a diversity of challenges and changes in research programs and perspectives. The era was marked by a dramatic increase in nuclear power plants operating nationwide, the Three-Mile Island (TMI) accident, a global energy crisis, and conflicting national energy priorities resulting from socioeconomic and political pressures. Government-sponsored research budgets waxed and waned to reflect diverse administrative policies and both President Carter's and Reagan's philosophical preferences.

This changing environment necessitated shifts in research and development (R&D) goals to satisfy urgent national needs and demanded application of innovative technology in both nuclear and nonnuclear areas. Thus, there was a flourish of activity and achievement in new and nontraditional modes as researchers searched for innovative ways to use nuclear power, ensure the safety and efficiency of nuclear reactors, develop alternative energy sources, and apply new technologies for thermal energy storage and national defense systems.

Government agencies also reflected the transition in research priorities and recognition of short- vs long-term technology development and application. Formed in 1974, the U.S. Nuclear Regulatory Commission (NRC) took over management of previous Atomic Energy Commission (AEC)-sponsored nuclear safety work. In 1975, AEC was renamed the Energy Research and Development Administration (ERDA) to emphasize its broader role. Only 2 years later, the U.S. Department of Energy (DOE) was established and emphasized a still yet more diversified role.

Along with the shift in program sponsorship from AEC to NRC and DOE, projects for the U.S. Department of Defense (DOD) and the National Aeronautics and Space Administration (NASA) dramatically increased. Within the division, each section had become its own marketing agent, and the success of these marketing efforts led to a variety of new sponsors and renewed emphasis on micromanagement. Because of the increased influence of outside authority, complex relationships developed between sponsors, Oak Ridge National Laboratory (ORNL), and division management. Rather than work on large research programs, effort was diversified into many smaller interrelated projects. Because of the interest in technology transfer, joint activities with subcontractors and private industry led to increased use of matrix management within the research environment.

Following reorganization and changes at the national level were those at the ORNL and the Division level. In 1977 the Reactor Division was renamed the Engineering Technology Division (ETD). Leadership within the division was also transferred. In 1978, G. G. Fee left the Division to assume a role in Central Management for the Y-12 Plant, and H. E. Trammell became Division Director. In 1984, Union Carbide Corporation terminated its role as manager of DOE plants; Martin Marietta Energy Systems assumed that role. Upon Trammell's retirement in 1989, J. E. Jones then returned to ETD to serve as its Director; W. R. Martin was named his Associate Director.

In September 1990, the division was recognized for achievement of a unique milestone. President C. C. Hopkins presented the Martin Marietta Distinguished Safety Performance Award to the employees of ETD in recognition of 40 years without a disabling injury. The division is on its way toward the next impressive goal—45 years of safe performance.

In 1992, ETD was composed of six research sections: Operational Performance Technology, Engineering Analysis, Applied Systems Technology, Thermal Systems Technology, Structural

Mechanics, and Pressure Vessel Technology. To reflect the current working environment more accurately, this chapter is organized by research section rather than by major programs, with one exception: the Space and Defense Technology Program is a relatively new initiative that exemplifies positive growth in a nontraditional mode for ETD. The great diversity of projects and programs described within each section attests to the breadth of ETD's research perspective and successful achievements during this era. Division organization charts for 1982 and 1992 are provided in Appendix B. Photographs of 1992 division members are shown in Appendix C.

## 5.1 OPERATIONAL PERFORMANCE TECHNOLOGY

In August 1991, the Nuclear Operations Analysis Center, previously one of ETD's six sections, was combined with the Performance Assurance Project Office, from within the Structural Mechanics Section, to form the Operational Performance Technology (OPT) Section. This newly created section combined OPT activities for NRC, DOE, and other sponsors and aligned resources and expertise in such areas as event assessments, performance indicators, data systems development, trends and patterns analyses, and nuclear standards. OPT also assumed the responsibility for management of the Quality Assurance Records Center for ETD and other organizations. This reorganization highlighted ORNL's significant role in assessing operational performance and indicated a commitment to further expand these activities.

Given the limited history of OPT, the history of its two predecessor organizations will be discussed individually to review their evolution and most recent activities before the merging of the two in 1991.

### 5.1.1 Nuclear Operations Analysis Center

During the late 1960s and 1970s, the Nuclear Safety Information Center (NSIC), under W. B. Cottrell's direction, continued to collect, abstract, and organize information relating to all aspects of nuclear safety. Computer files were established to replace the unmanageable 5- by 8-in. card filing system. The ORNL Computing Technology Center's program development for NSIC became the prototype and model for data bases by other ORNL information centers. In 1967, 10,700 items were already in the data base. Eventually the NSIC data base became part of the RECON system, operated by DOE's Office of Scientific and Technical Information.

NSIC's second major thrust was the preparation of technical analyses on specific topics. This continuing effort produced an important series of NSIC reports throughout the 1970s and early 1980s. Building on previous successes, such as the reactor containment handbook and indexed bibliography of accessions, NSIC established an impressive library on a broad range of topics.

A third continuing focus was publication of the journal *Nuclear Safety*. From its beginning as a small quarterly publication, subscriptions had grown to more than 2000 in 1975; the total print was 2000 copies above those for the paid subscriptions. This widely recognized, award-winning publication remained under the editorship of W. B. Cottrell, who combined this function with that of NSIC director. Early section editors for the journal included W. K. Ergen, H. B. Piper, M. L. Winton, R. L. Scott, and others.

The increasing number of nuclear plants in operation during the late 1970s and 1980s and, specifically, the TMI accident in 1979 had profound effects on NSIC. After TMI much more attention was paid to the operation of nuclear power plants rather than their design and construction, which had held center stage theretofore. This recognition was reflected by organizational changes within the NRC; the newly established Office for the Analysis and Evaluation of Operational Data became the chief supporter and sponsor of NSIC's work. In 1981 this change in emphasis was further reflected by the formation of the Nuclear

Operations Analysis Center (NOAC), the direct heir of NSIC; the latter then became one activity encompassed within NOAC. Upon W. B. Cottrell's retirement in 1984, management responsibilities were split: J. R. Buchanan took over management of NOAC, and E. G. Silver became editor of the *Nuclear Safety* journal.

NSIC continued to update its original data base until April 1984 when the inputting of all nuclear-safety-related reports was terminated; only the Licensee Event Reports (LERs) continued to be added. The analysis of operating events, as reported in LERs, became central to NOAC's work. NOAC undertook the in-depth analysis of these events at U.S. nuclear plants and collected the results in another large computerized data base, the Sequence Coding and Search System (SCSS). In 1984, N. M. Greene, G. T. Mays, and M. P. Johnson published a user's guide for the SCSS. W. P. Poore was also a significant contributor to this ongoing effort.

Other compilations of operating experience were also prepared during the mid-1980s. Another major undertaking, still ongoing in 1991, is the analysis of event sequences that might have but did not progress to major accidents. Such precursors to potential severe core damage accidents are analyzed, using probabilistic risk assessment methods to underpin the accident-probability estimates that constitute ultimate results of nuclear safety calculations.

Also in 1991, *Nuclear Safety* celebrated its 32nd anniversary. As sole survivor of the "rainbow series" of publications, *Nuclear Safety* continues to be a valuable resource for reactor designers, builders, and operators and for researchers, administrators, and safety officials in both government and private industry.

With the retirement of J. R. Buchanan in mid-1991, the section was again reorganized as the OPT Section. Under the leadership of G. T. Mays, it includes NOAC as one of its elements, with the *Nuclear Safety* journal as another, plus the new project office and the QA Records Center to complete its organization chart.

### 5.1.2 Performance Assurance Project Office

In 1967, the Nuclear Standards Program, which had been an ongoing activity at all Liquid-Metal Fast Breeder Reactor (LMFBR) Program sites, was established by the AEC as the Reactor Development and Technology (RDT) Standards Program. The program filled an urgent need to strengthen engineering practices in LMFBR contractor activities to ensure their success and the safe, reliable operation of important and valuable test and demonstration facilities. The program implemented systematic procedures to ensure that technical criteria, standards, codes, and requirements were used and that recognized standard practices were used or developed for use. The program was managed by ORNL Engineering under the RDT Standards Office, which coordinated activities with all LMFBR contractors, prepared RDT standards, provided project support, and actively participated in and supported national consensus standards (NCS) development efforts.

In 1977, DOE's Nuclear Power Development Division (NPD) became the focal point for standards activities within all its Nuclear Energy (DOE-NE) programs (except Naval Reactors). In 1978, the standards policy for NPD was issued. At the same time, the RDT Standards Office was moved from Engineering to ETD and renamed the Nuclear Standards Office (NSO) to reflect an intended broader coverage of nuclear programs. J. M. Corum was named manager of the overall standards program because of the related LMFBR High-Temperature Structural Design Program for which he was responsible. Later in 1978, ORNL was given broadened program management responsibilities according to an October 1978 Management Agreement between ORNL, NPD, and the DOE Oak Ridge Field Office. Through this management agreement, the Nuclear Standards Management Center (NSMC) was established at ORNL to handle all aspects of the expanded standards program. F. L. Hannon managed the NSO of the NSMC, while E. G. Silver managed outreach activities to promote nuclear standards activities.

The main objective of the NSMC was to ensure that information and experience gained in the course of programs funded by DOE-NE were documented in standards suitable for use in future programs and activities involving nuclear facilities in both the public and private sectors. To meet its objective, NSMC was involved in several technical activities across the DOE complex: preparing RDT Standards (later referred to as NE Standards) for review, publication, and update; managing working group meetings of DOE and DOE contractor representatives to discuss and resolve standards development and application issues; supporting DOE efforts to convert existing RDT/NE Standards to NCS in accordance with Office of Management and Budget Circular A119; assessing implementation of standards programs at selected DOE contractor sites; developing and maintaining data bases on the status of standards development, keywording, conversion activities, and personnel involved in standards development activities; disseminating program information to DOE and its contractor organizations, other U.S. government agencies, and private industry upon request; and conducting meetings with selected organizations to promote the development and use of standards.

J. M. Corum served as the manager of NSMC at its creation in 1978. H. L. Moseley joined NSMC in 1980 and currently serves as manager of the Performance Assurance Technical Staff. Other current employees who were part of the original NSMC organization are S. D. Jennings and F. C. Olden. C. A. Burchsted provided technical support to DOE standards efforts in the area of air cleaning until his untimely death. R. M. Fuller was also active until his retirement. For a brief period in 1981, J. N. Robinson managed NSMC; later that year W. L. Cooper, Jr., was named manager of NSMC and remained so for the next decade.

During Cooper's tenure, the role of NSMC evolved in response to the changing priorities of DOE. The downturn in the work on the DOE Liquid-Metal Reactor (LMR) Program served to deemphasize DOE's commitment (and concurrent NSMC funding) to the standards program. However, NSMC responsibilities involving the Unusual Occurrence Reporting (UOR) Program began to increase.

The UOR Program emerged from an RDT Standard prepared to standardize occurrence reporting within the DOE-NE programs. NSMC provided support to DOE through the review, analysis, and keywording of UORs for further analysis of occurrence trends; maintenance of a UOR data base; and preparation of trend reports and quarterly DOE management briefings on UOR trends. The standard was later converted to a DOE Order and implemented across all major DOE program areas. Under the new DOE Order, NSMC responsibilities expanded to support UOR data basing and trending in all program areas.

In recognition of its expanded role, NSMC was renamed the Performance Assurance Project Office (PAPO) by DOE in 1985. PAPO's primary focus was to provide technical and management support to the DOE Headquarters organizations (NE and Environment, Safety, and Health) on the RDT/NE Standards and UOR Programs. F. C. Zapp and R. C. Hudson supported ORNL and DOE efforts under the RDT/NE Standards Program before their retirements in 1988 and 1990, respectively.

In 1989, the DOE Office of New Production Reactors gave PAPO the "lead organization" assignment to evaluate applicable codes and standards issues. PAPO restaffing was initiated in response to the emerging DOE task areas. The new staff included D. L. Williams, Jr., named PAPO manager upon Cooper's retirement in 1990, T. W. Horning, and D. J. Spellman.

In 1990, work on the UOR Program was discontinued because of a change in DOE organizational responsibilities for occurrence reporting as implemented through DOE Order 5000.3A. In its place, authorization was received to assist in developing and implementing a DOE-wide Performance Indicator (PI) Program. PAPO's responsibilities under this program have included preparation of program guidance documents; development of program training materials and conduct of training for DOE and DOE contractor personnel; development of PC-based programs and methodologies for assimilating performance data and preparing management reports to summarize and qualify performance; and preparation of DOE summary-level PI reports for approval by the Assistant Secretary for NE and submission to the Secretary of Energy. Through this program, PAPO has continued its historical role in providing technical support in the analysis and trending of events in the DOE complex to support management efforts in achieving continuing improvement in all phases of its operations.

In 1991, PAPO work for the RDT/NE Standards Program changed to new support activities for the DOE Standards Program. The updated emphasis on standards resulted from identified needs for standards in certain DOE program areas and DOE's subsequent revision of Order 1300.2 (issued as Order 1300.2A), which describes DOE's policy on the standards' development and application. Using the expertise and experience gained through the RDT/NE Standards Program, PAPO is assisting DOE-NE in developing the required program guidance and information resources to successfully implement the DOE Standards Program. Also, in 1991, PAPO began separate efforts to provide technical assistance to Energy Systems in the implementation of the new Occurrence Reporting procedures mandated by DOE Order 5000.3A and the identification of "Lessons Learned" from operating experience gained by government and commercial organizations other than Energy Systems for inclusion in the Energy Systems Lessons Learned System. Both initiatives have served to increase Energy Systems' awareness of PAPO's capabilities.

Since the creation of NSMC/PAPO, numerous ORNL personnel have continued to provide support and expertise to the various programs assigned to this office. The blend of capable existing staff with new experienced personnel has served to reestablish PAPO as a "center of excellence" for DOE and DOE contractor organizations on issues related to the continuing improvement of operational performance through the development and maintenance of standards; compliance with mandatory occurrence reporting requirements; and the identification, analysis, and reporting of facility performance data.

## 5.2 ENGINEERING ANALYSIS

From 1975 to 1991, the Engineering Analysis Section responded to ever-growing demands for R&D beyond its traditional role of nuclear research. In addition to continuing studies of advanced reactor concepts and utilization of nuclear power, various alternative energy development areas became integral to the section's success. The broadening of expertise and experience reflected transitions within the political and socioeconomic environments during this time.

In 1975 the Engineering Analysis Section was involved in a diverse set of interrelated studies to promote the use of nuclear power to displace the use of oil and gas. I. Spiewak, the section head, and J. E. Jones led one program focused on the potential of nuclear power to provide process heat for industrial applications such as steelmaking, oil refining, coal gasification and liquefaction, processing of oil shale and tar sands, and the hydrogen extraction from coal and water. T. D. Anderson and O. H. Klepper studied the feasibility of using nuclear reactors at industrial sites—large reactors to supply multiple users and small modular reactors to serve individual users. At that time they also investigated the feasibility of establishing very large power parks with as many as 40 nuclear reactors.

From 1976 to 1982 a study of conceptual Nuclear Energy Centers (NECs) was managed for DOE by T. E. Cole and H. F. Bauman. This study was directed to the feasibility and practicality of developing NECs, with 9 to 12 large reactors [1250 MW(e)] each, at specific sites to be determined in the Southeastern and Western United States and to determine differences due to location, if any. Two specific sites were studied in cooperation with the states involved, South Carolina and Utah. To address the questions of feasibility and practicality and to determine significant regional differences, technical, socioeconomic, environmental, radiological, and institutional issues were addressed. For each site the concept was found to be feasible, but further analysis of institutional and socioeconomic issues would be required before practicality could be resolved. Major differences were found between the two sites in almost every aspect.

The Studies and Evaluations Program, managed first by L. L. Bennett followed by H. I. Bowers, began a series of economic evaluations of these concepts and began building cost models and computerized cost-estimating programs. The program also began considering nonnuclear concepts such as fluidized-bed combustion of coal and energy conservation.

With the election of Jimmy Carter in 1976, the complexion of the research changed significantly. The section became involved in conservation, solar, and fossil energy development areas. J. C. Moyers and E. C. Hise developed the Annual Cycle Energy System that provides space heating and cooling by means of a heat pump. In winter the heat pump produces ice that is used to cool in the summer, gaining a great reduction in energy use. This research project won the National Society of Professional Engineers (NSPE) Outstanding Engineering Achievement Award. In a collaborative effort with the Environmental Sciences Division, the section was working to develop beneficial uses for the rejected heat from power plants. M. Olszewski developed concepts to use this otherwise wasted heat to grow tomatoes and to raise fish. T. D. Anderson led a study to evaluate the proper role for solar energy for producing electricity. He determined that the solar collectors would have to be cheaper than billboards to be economical for this application and that scenario was unlikely in the near future—a prophesy that has come true.

About this same time the section became involved in coal technology. A. P. Fraas had developed a concept to use a fluidized-bed coal combustor to provide process heat and electricity. E. C. Fox and R. L. Graves examined new ways to increase the coal use in industry and studied advanced combustion systems for steam and electricity production. D. M. Eissenberg invented a new process for separating pyrites and ash from coal using a strong magnetic field. This technology was patented and later won an IR 100 award.

Also with the Carter Administration came concern over the ability of other countries and groups to divert nuclear material from power reactors for weapons production. This issue was addressed through the DOE Nonproliferation Alternative Systems Assessment Program established at ETD and led by I. Spiewak and A. J. Frankel. Engineering Analysis staff also researched different fuel cycle options from both technical and economic viewpoints.

District heating was recognized as an excellent way to use the economy of scale of a large power plant in a very efficient manner by providing thermal energy directly to commercial, industrial, and residential customers. M. A. Karnitz and I. Spiewak initiated a cooperative study with the city of Minneapolis and Northern States Power to evaluate the feasibility of building a large hot water district heating system. Several communities in the northern United States considered building such a system because of this effort. H. I. Bowers and M. A. Kuliasha also were involved in electrical load management and cogeneration.

In 1979 J. E. Jones began a program for the Tennessee Valley Authority (TVA) to provide technical assistance in development of fluidized-bed combustion technology. For several years, bench-scale combustion tests, coal-feeding experiments, hydrodynamic modeling, and technical and economic evaluations were conducted. The results from this work were used directly in the design of the TVA Pilot Plant.

By 1979 the Fossil Program had grown substantially, and a separate section was formed. J. E. Jones was the new section head with D. M. Eissenberg, E. C. Hise, and D. W. Burton as group leaders. E. C. Fox headed up the TVA program, R. L. Graves was developing new advanced fluidized-bed concepts, and R. S. Holcomb led the coal combustor cogeneration development program.

W. L. Greenstreet, R. L. Carmichael, E. L. Churnetski, and M. L. Myers supported the DOE Economic Regulatory Administration in steps taken to implement the Fuel Use Act passed by Congress. The purpose of this act was to convert electrical power plants from use of oil and gas as primary fuels. ORNL conducted engineering and environmental impact studies and prepared cost estimates to assess the feasibility of conversion.

The magnetic separation of coal grew into a multiproject coal cleaning program. J. C. Moyers researched the automation of conventional coal cleaning plants and analyzed coal cleaning systems for TVA and DOE, and A. S. Holman was developing computer models that would optimize the operation of coal preparation plants. D. M. Eissenberg was responsible for a program on processing system components. W. L. Greenstreet led a project for determining R&D needs for critical components and preparing program plans to meet these needs. These components were to withstand the hostile environments and meet stringent demands associated with coal gasification and liquifaction processes. The components addressed were slurry pumps, compressors, and expanders (M. L. Lackey); valves (W. K. Kahl); heat exchangers (E. L. Churnetski); and coal preparation equipment (J. R. Horton, Engineering).

In a parallel effort, long-term support for the Office of Energy Research began. J. P. Nichols led a series of technical and economic assessments of alternative energy sources and later conducted reviews of the DOE research programs. I. Spiewak was to direct all support activities to the Energy Research Advisory Board (ERAB), DOE's standing review board.

Because the TMI accident irrevocably altered the course of nuclear power, the focus of the section's research programs changed. The NRC asked ORNL and ETD to develop a capability to analyze accidents in pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) up to and beyond core melting, to understand the complex interactions throughout the plant during the course of the accident, to evaluate possible operator actions, and to develop a basis for possible plant improvements. S. A. Hodge, S. R. Greene, C. R. Hyman, and L. J. Ott were pioneers in this research, which is still one of the NRC's foremost research areas.

With the Reagan Administration drastic changes were implemented in the priority and conduct of DOE research. Emphasis was placed on the private sector for technologies that use fossil fuels. As a result, the funding for fossil research at national laboratories was severely curtailed.

At this time I. Spiewak retired, and C. D. West took over support activities for ERAB. The Engineering Analysis Section was combined with the Fossil Energy Section with J. E. Jones as the head.

Consensus from program planning sessions in 1981 was that one of the most important research topics was the use of energy in the transportation sector (the principal use of oil). By 1982 R. L. Graves was working with DOE to help prepare its Heavy Duty Transportation Program Plan. In 1983 the section was selected to manage the DOE Alternative Fuels Utilization Program. The program goal was to perform the basic R&D to provide the nation with a selection of technologies that used nonpetroleum fuels for highway transportation. A fuels and combustion laboratory was built, and a variety of experimental engines were used to test and evaluate fuels and materials. By 1986 the program had expanded to include the national demonstration of methanol fuel technology. R. N. McGill was managing fleets of test vehicles at Lawrence Livermore, Argonne, and Oak Ridge National Laboratory.

The aging of nuclear power plants became an important research issue as the bulk of the nation's reactors reached maturity. D. M. Eissenberg led the NRC's Nuclear Plant Aging Research Program; its objectives are to identify and evaluate practical methods for detecting, monitoring, and assessing the severity of time-dependent degradation of electrical and mechanical components in plant safety systems. The emphasis was placed on the evaluation of techniques to detect the onset of incipient defects before failure and the need for maintenance to mitigate these defects. Reports on failure modes and causes, measurable parameters for diagnoses, and monitoring methods for nuclear plant equipment were prepared by J. C. Moyers (air compressors, dryers, and heat exchangers); D. A. Casada (auxiliary feedwater systems); W. L. Greenstreet, G. A. Murphy, and others on motor-operated valves, check and other valves, pumps, and other safety-related equipment. The success of this program is partially evident from a list of inventions by D. M. Eissenberg, H. D. Haynes, and D. A. Casada. These inventions, licensed to private companies, include motor current signature analysis to evaluate the condition of any motor-driven component and magnetic signature methods for nonintrusively monitoring the condition of check valves.

As a part of the program on aging, W. L. Greenstreet was a member of the American Society of Mechanical Engineers (ASME) Committee on Operation and Maintenance of Nuclear Plants. The purpose of this organization is to develop codes and standards for operation and maintenance of critical components in nuclear plant systems. Greenstreet was instrumental in the publication of the first *ASME Code on Operation and Maintenance of Nuclear Power Plants.*

In parallel efforts during 1984, the division began a series of studies for potential designs of a reactor to replace the High-Flux Isotope Reactor (HFIR). This reactor, then called HFIR II, later the Center for Neutron Research, and finally the Advanced Neutron Source (ANS), was to be the finest research reactor in the world with the highest flux of neutrons and extensive facilities specifically designed for research using neutrons.

The Strategic Defense Initiative (SDI) became a significant national program as a part of the U.S. Cold War effort during the 1980s, which focused on early detection and in-flight destruction of enemy strategic missiles. Recognizing the need for large electric power sources for space-based defense platforms, a group led by J. E. Jones and including J. P. Nichols and A. P. Fraas (then a consultant to the division) proposed development of multimegawatt power systems that could provide electricity for station maintenance and weapons systems on orbiting platforms. In the resulting ORNL Multimegawatt Program, led by J. P. Nichols with R. S. Holcomb and J. C. Moyers as major participants, concepts were developed based on potassium vapor Rankine power systems driven by either liquid-lithium-cooled or boiling-potassium-cooled reactors. Although other concepts were proposed and developed by other national laboratories, the alkali-metal-cooled, potassium-vapor power cycle concept is recognized as the leader in terms of long life and low system mass. The program faltered near the end of the decade from lack of funding due to SDI emphasis on weapons development, rather than power system development, and changes in SDI defense concepts.

In 1985 J. E. Jones left the division to head ORNL's Reactor Program Office. W. G. Craddick was then named section head. C. D. West and the Irradiation Engineering Group were moved into the Engineering Analysis Section in 1986. The Irradiation Engineering activity is the longest continuous experimental activity in the division. For over 30 years this group has designed, fabricated, and operated the irradiation experiments conducted in the Laboratory's research reactors. These experiments have provided much of the basic data used to evaluate the irradiation damage of nuclear reactor pressure vessels (RPVs), integrity of the fuels, candidate material for the first wall of a fusion reactor, and many other nuclear-related material radiation damage issues. In 1987, C. D. West left to manage the ANS effort, which became a major ORNL program. K. R. Thoms assumed leadership for the Irradiation Engineering Group.

In 1987 the HFIR was shut down because of concern over the embrittlement of the RPV. Division staff were recruited to perform the analysis and studies needed to restart the HFIR. W. G. Craddick left to manage the Reactor Technology Section in the newly formed Research Reactors Division, and E. C. Fox replaced him as head of the Engineering Analysis Section.

Engineering Analysis staff continued to be involved in the development of advanced reactor concepts. J. P. Sanders, J. C. Cleveland, and J. C. Conklin have extensively evaluated the safety aspects of the High-Temperature Gas-Cooled Reactor (HTGR). J. C. Cleveland proposed, helped design, and evaluated the first and only planned loss-of-coolant accident (LOCA) in an operating reactor. At the AVR (HTGR) in Germany, the helium circulators cooling the reactor were shut down. When the reactor was cooled through its inherent natural circulation features, the fuel temperatures stayed well below the temperature at which fission products would be released, thus demonstrating the inherent safety of this concept. In 1990 H. T. Kerr began an initiative to evaluate and develop a direct-cycle modular high-temperature gas-cooled reactor (MHTGR). This system, which uses helium as the working fluid in a Brayton cycle gas turbine, is expected to have all of the attractive safety features of the steam-cycle MHTGR but be simpler and cheaper, with a higher thermal efficiency.

In 1989, DOE decided that a New Production Reactor (NPR) was needed to replace the aging reactors at Savannah River. The Engineering Economic Evaluations Group, headed by C. R. Hudson, was chosen to evaluate the economic claims of the proponents of all of the various proposed reactor systems [light-water reactor (LWR), heavy-water reactor (HWR), LMR, MHTGR, and various accelerator designs]. Hudson, K. A. Williams, L. C. Fuller, R. L. Reid, and B. Cowell were asked, first, to evaluate and to establish credible estimates for the proposed concepts, then, to analyze the reasonable cost from the two designs specified for the final selection process, and finally, to establish the basis for the NPR project budget that was sent to Congress.

## 5.3 APPLIED SYSTEMS TECHNOLOGY

The seeds for the present-day Applied Systems Technology (AST) Section resided within several earlier Reactor Division/ETD sections. AST primarily evolved from the Fast Reactor Safety and Core Systems (FRS&C) Section and the Experimental Engineering Section. See Table A.1 for listing of sections. There were also contributions from the Fossil Energy and the Thermal Systems Technology Sections. Some major milestones along the way were as follows.

In 1975, M. H. Fontana was head of the FRS&C Section, and R. E. MacPherson led the Experimental Engineering Section. In 1977 the FRS&C Section changed its name to the Advanced Reactor Systems Section. In 1980, the name was again changed to the Advanced Concepts Development (ACD) Section.

From 1975 to 1982 under M. H. Fontana's leadership, the section concentrated on nuclear-safety-related programs for both DOE and NRC. Programs sponsored by DOE consisted primarily of two tasks; these embraced work in connection with the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility (formerly FFM) and the Core Flow Test Loop (CFTL). The THORS Facility was managed by J. L. Wantland and operated by B. H. Montgomery (Experimental Engineering Section), R. H. Morris, and J. J. Carbajo. The THORS Program investigated core thermal hydraulics, effects of channel blockages, and boiling sodium behavior under accident conditions associated with the LMFBR. The CFTL was developed by U. Gat and run by J. P. Sanders; it was designed to study steady-state and transient behaviors of fuel elements for the gas-cooled fast reactor (GCFR).

CFTL construction was completed in September and shakedown tests were concluded in December 1981. Also in 1981, the GCFR project at General Atomic was canceled, and the name of the Core Flow Test Facility was changed to Component Flow Test Loop. In late 1983 and early 1984, a preliminary, core-support, performance test was conducted as a part of the HTGR studies. The test was defined by W. P. Eatherly, Metals and Ceramics Division. J. P. Sanders of the Reactor Division was responsible for test preparation and execution; he was aided by U. Gat, H. C. Young, W. R. Huntley, and others. The purpose was to examine the effect of oxidation on stresses in this support structure, which was composed of graphite posts, or columns, supporting a load-bearing structure upon which the reactor core was to be mounted. The stresses of interest were those at the interface between the spherical upper end of each column and the spherical socket of the mating structural component. Since changes in the mating surfaces due to oxidation could have significant deleterious effects on stresses in the members, and, hence, their useful lives, it was important to examine this phenomenon.

Because graphite to be used in the reactor was unavailable at the time of the test, a substitute graphite was used. During operation, helium was circulated in the loop, with the pressure and temperature being 1050 psi and 1290°F, respectively. Although the test was successful, cancellation of this work due to lack of funds precluded follow-up testing.

Work under the Aerosol Release and Transport (ART) Program for NRC was led by T. S. Kress and included efforts by R. E. Adams, L. F. Parsly, A. L. Wright, A. W. Longest (Experimental Engineering Section), H. W. Bertini, J. S. White, M. L. Tobias, and others. The ART Program developed a substantial data base and code validation for the behavior of fuel and fission-product aerosols under accident conditions for both LMFBRs and LWRs. NRC also sponsored substantial safety analysis work on the HTGR. In 1980, Fontana started the Severe Accident Sequence Assessment Program, which was the forerunner of both the major Industry Degraded Core Rulemaking (IDCOR) Program and the BWR Severe Accident Technology Program, now under S. A. Hodge.

During this period, the Experimental Engineering Section efforts included evaluating coal combustion technology, providing experimental support to the nuclear safety programs, and studying the thermodynamics of alkali metal vapor cycles. D. B. Lloyd was a major contributor to these programs; others included W. R. Huntley, D. L. Clark, and R. E. Helms. From 1978 to 1982, the section's budget declined because of completion of the CFTL and transfer of major coal programs into the newly formed Fossil Energy Technology Section.

In 1982, M. H. Fontana gave up leadership of the ACD Section and left ORNL to develop the IDCOR Program, which was funded by all nuclear utility companies in the United States to examine severe accident behavior of LWRs. After about a year, during which time the section was held together jointly by T. S. Kress and J. L. Wantland, the ACD Section was folded into the Experimental Engineering Section under management of R. E. MacPherson.

From 1982 to 1986, the Experimental Engineering Section continued to emphasize heavy experimental work related to THORS, ART, and GCFR. Funding from NRC and DOE increased slightly during this period. These three programs made up about 95% of the section's budget in 1982 but had fallen to about 50% in 1986. At that time, the other half of the budget was made up of a diverse set of much smaller projects, many sponsored by DOD and related to military site environmental problems. The Fuels and Combustion Program, with its Alternative Fuels Utilization Program and Methanol Fleet, also came into existence at the end of this time period under the leadership of R. L. Graves and R. N. McGill.

In 1985 R. E. MacPherson retired, and D. W. Burton was named the new section head for Experimental Engineering. In 1986 under E. C. Fox, the Energy Systems Technology Group (which addressed studies on hazardous waste technology, alternative fuels for power and transportation, fossil energy uses, and reliability analyses) was transferred from the Engineering Analysis Section into the Experimental Engineering Section in a swap for the Materials and Systems Technology Group, which was transferred into Engineering Analysis. The latter group was under C. D. West and embraced irradiation engineering activities and early work on the ANS to replace the HFIR. In 1987, the Experimental Engineering Section was given its present name, Applied Systems Technology (AST) Section.

As of 1992, the AST section consists of four groups: (1) Process Systems Technology, managed by R. M. Schilling; (2) Fuels, Combustion, and Engine Technology, led by R. L. Graves; (3) Energy and Nuclear Sciences, headed by U. Gat; and (4) Passive Countermeasures, directed by M. A. Akerman. The first three were formed in 1987, with the first and second being from remnants of the Fossil Energy Section and the third from remnants of both the Experimental Engineering Section and the ACD Section. The roles of the four groups are as follows.

The Process Systems Technology Group activities are focused on hazardous waste minimization, recycle, and destruction technologies; combustion systems evaluation and testing; system risk and reliability analyses; and chaos methodology. Members of this group include S. M. Crosley, C. S. Daw, J. M. Hoegler, R. P. Wichner, D. B. Lloyd, J. F. Thomas, R. H. Staunton, M. L. Tobias, V. K. Wilkinson, and J. M. Young.

The Fuels, Combustion, and Engine Technology Group is engaged in advanced diesel engine technology R&D embracing fuels and materials of construction. Members of this group are R. N. McGill, B. H. West, J. C. Conklin, N. Domingo, and R. P. Krishnan.

The Energy and Nuclear Science Group addresses ideas and projects for advanced energy sources through experimental engineering work. Reactor systems now being considered include molten salt and safe reactor concepts. J. P. Sanders is a member of this group.

The Passive Countermeasures Group was transferred from the Thermal Systems Technology Section in 1989. Currently its primary activity is to develop materials and components for shielding Army tanks and other equipment against hypervelocity and ballistic impact, blast, and laser threats. This group was formerly under D. G. Thomas, with J. E. Smith being a member.

From 1986 to 1992, the AST programs remained as a diversified set of many, somewhat disconnected, projects funded by DOE, DOD, and NRC. These projects involved R&D for fuels, engines, and combustion technology; facility and nuclear safety analyses; environmental problems at various DOD sites; and armor-plating technology. AST funding peaked in 1989, but it later declined because DOE transferred the major Alternative Fuels Utilization Program to the Solar Energy Research Institute in 1989, and the research staff completed some of the major DOD projects. In 1991, D. W. Burton retired, and T. S. Kress was named to lead the AST Section.

## 5.4 THERMAL SYSTEMS TECHNOLOGY

The history of the Thermal Systems Technology Section from 1975 to the present is a microcosm of ETD's history—a period of transition from a time when expertise was applied mostly to reactor systems to a time when it is applied across a broader spectrum. Through most of this period H. W. Hoffman led the section, which was known as the Heat Transfer–Fluid Dynamics Section until late 1982. Its mission statement in 1975 specified its work as focusing on "energy deriving from nuclear fission sources."

The expansion in scope had begun even then. In the midst of a section mainly focused on work related to nuclear energy, S. L. Milora, S. K. Combs, and others were working on "heat utilization," which included "low-temperature cycles, ocean thermal gradient, and thermal storage." The latter was the beginning of a long and fruitful effort for the section. Efforts in this area have grown and shrunk with shifting national priorities but still continue today at a significant funding level.

The work in nonnuclear energy applications was broader than just thermal energy storage (TES), though that portion has demonstrated the most longevity. The combination of sharply increasing oil prices and President Carter's opposition to nuclear fuel reprocessing contributed to the government's increasing interest and expenditures in alternative energy sources. In the late 1970s, the section conducted significant research efforts in geothermal and ocean thermal energy generation, led by R. W. Murphy, and in atmospheric thermal effects, led by A. A. N. Patrinos. The work in the use of naturally occurring thermal gradients for energy storage included the experimental determination of the relevant physical properties of various working fluids, determination of system efficiencies, and investigation of design options for power-generating systems. Much of this work centered around evaluation and enhancement of heat exchange technology; for example, the use of fluted tubes was found to enhance heat transfer and condensation. N. Domingo and C. V. Hardin contributed to this effort for several years.

Work in atmospheric thermal effects was aimed at determining the nature and extent of changes in the local weather pattern caused by the presence of power plants. Studies of the 3160-MW(e) fossil-fueled Bowen Plant, operated by Georgia Power, led to the conclusion that the heat releases from the plant affected the distribution but not the overall quantity of precipitation in the region around the plant. N. C. J. Chen worked for several years with Patrinos on this effort.

As the work in geothermal and ocean thermal energy generation and atmospheric thermal effects progressed in the late 1970s, so did the work in TES. While experiencing somewhat slower growth, this effort continues to this day. In its early years, the one-man effort, conducted by R. J. Kedl, included investigation of techniques using form-stable polyethylene, liquid desiccants, and immiscible fluids. In 1978 D. M. Eissenberg assumed overall leadership for the alternative energy technologies and for the TES effort in particular. The research expanded to begin investigating concepts for diurnal and industrial energy storage, an area of continuing research.

By 1980 J. F. Martin assumed leadership for the TES effort. Investigations had broadened, particularly in the area of phase change energy storage, a technology that is still an active area of research. The early 1980s also saw the arrival of several staff members who would play important roles in the division for many years, including M. Olszewski, J. J. Tomlinson, and R. N. McGill. The work continued to expand to include analysis of residential, commercial, and industrial energy use patterns and mechanical energy storage via flywheels. T. K. Stovall and L. Jung joined the expanding research.

The TES activities were focused on two major application areas: reuse of industrial reject heat, headed by M. Olszewski, and heating and cooling of commercial and residential buildings, headed by J. J. Tomlinson. The industrial reject heat portion of the program sought to increase the thermal efficiency of batch processes and began demonstration projects in the food, aluminum, and brick industries. The recovered thermal energy was stored via conventional means (primarily as sensible heat) and reused for industrial processes and district heating for cities. These projects were all near the implementation phase (actual hardware was to be built and installed) when in 1980 the focus of the DOE program underwent a radical change. During the Carter presidency the DOE Energy Storage Program focused on demonstrating existing technology in novel applications. Under the Reagan Administration DOE focused on long-term, high-risk, high-payoff development efforts. Thus the industrial TES Program underwent a major shift in emphasis in 1981. The demonstration projects were terminated, and research began on innovative TES technologies for capture, storage, and reuse of industrial waste heat.

In response to the energy crises of the 1980s, the staff began research to develop TES systems for use in residential and commercial buildings. Use of solar energy for building heating or cooling requires storage to extend the solar resource to nighttime periods. Through storage, baseload coal or nuclear power plants could provide electricity for building space heating or cooling to offset the need for electricity generated using oil or gas. Therefore, the case for TES was strengthened because it provided a way to increase the use of renewable energy and to reduce the need for peak electrical power derived from oil and gas. While much of the research to develop advanced TES technologies for buildings was managed for DOE through ETD and conducted through subcontracts, several rather large TES experiments were conducted in Building 9204-1.

One such experimental facility, the Thermal Energy Storage Test (TEST) facility, was designed to test latent heat storage system prototypes produced by independent manufacturers for use in residential and small commercial building heating and cooling applications. These systems consisted of tanks filled with hydrated salts and a heat exchanger. Selected were hydrated salts that froze or melted energetically in a temperature range suited for space heating or cooling. M. P. Ternes designed much of the TEST loop, D. J. Fraysier coordinated its construction, and J. F. Thomas conducted some early tests on prototypical systems. J. J. Tomlinson designed an experimental facility, operated by M. P. Ternes and J. J. Carbajo, to examine clathrates (binary icelike structures) as potential cool storage media. This team discovered and patented methods for tailoring the melting temperature of the clathrate and enhancing the rate of formation during the freezing half cycle.

The use of thermal energy from the ground beneath the crawl space of a house for preheating or precooling the air to the outside unit of a heat pump was examined in a field experiment conducted by R. N. McGill, M. P. Ternes, and D. J. Fraysier in Karns, Tennessee (a small community nearby). This facility consisted of three outwardly identical houses: one heated and cooled by a conventional heat pump, one in which outside air is drawn through the crawl space before passing through the heat pump, and the other in which air from the outdoor unit is recirculated in the crawl space. These experiments, supported by DOE and the Electric Power Research Institute (EPRI), proved the validity of the concept and estimated the energy savings possible in various regions of the country.

Experimental TES work in the division grew significantly with the design and development of the Ice Storage Test Facility. This facility, supported through EPRI in a Work-for-Others program, was designed to determine the performance characteristics of commercial ice storage systems being used for off-peak commercial building cooling and to work with manufacturers to improve system designs and performance. The facility consists of a large, highly instrumented built-up refrigeration and heat rejection system for testing various methods for making ice. T. K. Stovall has tested seven ice storage units along with one unit based on a material that melts at 41°F. Several manufacturers have applied results from these tests to improve the efficiencies of their designs. Because of its experience with cool storage for commercial buildings, the section was asked to design and monitor the performance of ice storage systems for the U.S. Army. J. J. Tomlinson prepared three different system designs for a Post Exchange at Fort Stewart, a barracks at the Yuma Proving Ground, and a Dental Clinic at Fort Bliss. The Army built and commissioned these systems, and R. J. Kedl measured their performance and reported to the sponsor. This information is now being used to prepare design guidelines for implementation of cool storage at Army facilities.

More recently, the diurnal TES Program has focused on developing plasterboard (wallboard) that contains a phase change material (PCM) for added thermal capacity. The PCM changes phase (melts or freezes) at 70°F , thus absorbing energy when the room temperature rises above 70°F (acting to cool the room) and releasing energy when the room falls below 70°F (heating the room). The plasterboard has a covering corresponding to that of drywall plasterboard. Analytical work has shown that the wallboard can reduce the supplemental heating requirement of a passive solar building by as much as 20%; further, this work has determined the optimal quantity of PCM needed. This analysis was made possible by the development of computer simulations of the freezing and melting behavior of the PCM wallboard and validation experiments conducted in a thermal testing fixture located at ORNL. R. J. Kedl designed, constructed, and used a small in-house facility to prepare full-scale, 4- by 8-ft sheets of plasterboard for field testing. Development of the PCM wallboard is continuing, and a major U.S. manufacturer of gypsum products is sharing the cost.

A second element was added to the energy storage efforts within ETD when management responsibilities for the Mechanical Energy Storage Technology (MEST) Program were transferred to ORNL in 1982. M. Olszewski and R. Steele conducted this program that focused on development of flywheel technology and examination of elastomeric concepts.* The flywheel testing facility was the most advanced in the country, particularly the instrumentation developed to detect incipient failure. This program continued through 1984 when DOE again reorganized their energy storage activities, because of declining budgets, and the MEST Program was terminated.

In 1985 space power applications were added to further expand the scope of TES work. ORNL was designated as the lead laboratory for the energy storage work within the DOE Multimegawatt Space Power Program. M. Siman-Tov led this effort, which focused on fuel cells (subcontracted to Argonne) and thermal and mechanical energy storage for sprint power† applications. At the same time, the staff began TES projects for sprint power needs in the SDI architecture, as well as for advanced solar dynamic power systems‡ being developed by NASA for Space Station Freedom. The TES work for SDI concentrated on developing a high-specific-energy TES system that could be used in the heat rejection system for sprint power systems. By storing the reject heat during the relatively short period of power generation and rejecting the heat over the entire orbit, substantial savings in mass and volume are possible for the heat rejection system. This program, conducted by M. Siman-Tov, developed a TES concept using lithium hydride as the PCM. The resultant energy storage system had a specific energy an order of magnitude larger than any previous system.

M. Olszewski managed a program to develop a TES system for NASA's solar dynamic receiver;§ objectives were to improve the thermal response

*Concepts involving material that can be twisted to store energy (e.g., rubber band).

†Sprint power—system that is called on periodically for high levels of power for short periods.

‡Solar dynamic—power system that uses solar energy and has rotating machinery in power conversion cycle.

§Solar dynamic receiver—receiver that hooks up to cycle above.

of the system and to increase the specific energy. The concept developed used a metal PCM* encapsulated in a lightweight containment (graphite, silicon carbide, or boron nitride). Using metallic PCMs resulted in thermal conductivities that were an order of magnitude higher than the baseline salt systems. Prototypical storage elements using germanium as the PCM with a graphite container were fabricated and successfully tested.

Modeling of the phase change process has also been an interest of the section. NASA funded a program, managed by M. Olszewski, to develop a unique model for performance analysis of phase change TES systems under microgravity conditions. The model is three-dimensional; accounts for conduction, convection, and radiation heat transfer modes; and includes volume changes and void growth due to solidification or liquefaction of the PCM. The truly unique feature of the code is that it includes the effects of void movement on the thermal profiles within the PCM. Void movement in a normal gravity environment is due to buoyancy, while Marangoni forces† dominate in a microgravity environment.

The second major area of activity for the section during the period from 1975 forward was analysis and experimentation in heat transfer and fluid flow related to nuclear applications and specific issues connected with the safety of commercial nuclear reactors. The NRC's PWR Blowdown Heat Transfer (BDHT) Separate Effects Program constituted the largest portion of this effort. D. G. Thomas headed the program for several years; subsequently J. D. White and then W. G. Craddick assumed leadership. The first isothermal blowdown test was conducted in February 1975 in the Thermal Hydraulic Test Facility (THTF). The first tests in the THTF were directed at investigating the sequences of events that might occur during the blowdown phase (first 20 to 30 s) of a postulated reactor LOCA. In later years the facility was used to obtain data that served as the basis for assessing and developing new heat transfer correlations for the rod bundle geometries relevant to commercial reactors. This work was sufficiently significant to cause NRC to revise the portion of 10 CFR 50 that prescribes the rules to be used in analyzing postulated accidents.

While the PWR-BDHT Program was the largest single program, it was not the only program in this area within the section. The Multi-Rod Burst Test Program, led by R. H. Chapman from 1974 to 1982, investigated experimentally the deformation and rupture during postulated accident conditions of the Zircaloy cladding used on commercial reactor fuel rods. Two major programs in the area of advanced two-phase instrumentation development were under the overall direction of D. G. Thomas. The Advanced Instrumentation for Reflood Studies Program, headed by P. A. Jallouk, and the Instrument Development Loop Program, headed by S. K. Combs, developed and tested advanced concepts for instrumentation to be used in investigating reactor safety both in this country and internationally. J. E. Hardy was another important contributor to both of these programs.

The commercial reactor safety programs not only produced significant technical results, but they also brought several people who continue to be important contributors to the division including D. K. Felde, D. J. Fraysier, C. R. Hyman, D. G. Morris, L. J. Ott, J. J. Robinson, and G. L. Yoder. In addition to benefiting from the efforts of talented new people, these programs received important contributions from long-time ETD staffers such as L. Jung.

In the early to mid-1980s there was a transition between the substantial thermal hydraulic efforts in support of NRC programs and the similar substantial efforts currently in progress to support research reactors. During this period the section provided support to a variety of programs. W. G. Craddick, D. G. Morris, and A. Sozer, in turn, provided support to the NRC's Safety Implications

*Metal PCM—germanium was the one developed; others were silica-containing alloys.

†Marangoni forces—forces occurring on a full surface due to surface tension variations across the surface.

of Control Systems Project being led by the Instrumentation and Controls Division. T. M. Anklam, W. G. Craddick, D. G. Morris, and C. B. Mullins provided heat transfer support to the Atomic Vapor Laser Isotope Separation (AVLIS) Program for enriching uranium. D. K. Felde spent some time on assignment as an NRC licensing examiner for commercial power reactors before also joining in to support the AVLIS Program. D. G. Thomas, J. E. Hardy, and G. L. Yoder supported space exploration and utilization through the analysis of various candidate space power cycles, the study of microgravity two-phase flow phenomena (including design of an experiment that was planned to fly on a space shuttle flight until the hiatus in shuttle flights caused by the Challenger accident), and the development of very lightweight but highly effective shielding for spacecraft against debris or kinetic energy weapons.

In 1987 the section began applying its thermal hydraulic expertise to research reactors, supporting both the HFIR restart efforts* and the design effort for the new ANS. The former effort was led by Morris; the latter, by Yoder. The HFIR support effort included analysis of issues requiring resolution in order to win DOE approval for restart, the most notable of which was the analysis of the HFIR's decay heat removal capability, and which has today progressed to analysis needed for the updated HFIR Safety Analysis Report. The ANS support effort includes responsibility for all thermal hydraulic analysis and experimentation in support both of ANS design and safety. The ANS thermal hydraulic support effort has relied on existing section staff—N. C. Chen, D. K. Felde, and M. Siman-Tov—as well as bringing new people into the section—A. E. Ruggles.

In 1989 a division reorganization added the Severe Accident Analysis Group, led by S. R. Greene, to the Thermal System Technology Section, which was now under W. G. Craddick's leadership. This group added to the level of support for HFIR and ANS, providing significant severe accident analysis support to both projects, as well as doing work for the NRC and, most recently, for the Savannah River reactors. The group includes S. E. Fisher, S. H. Kim, R. H. Morris, D. B. Simpson, and R. P. Taleyarkhan. This same reorganization returned A. Sozer to the section; he is providing additional thermal hydraulic support to the HFIR.

## 5.5 STRUCTURAL MECHANICS

The Structural Mechanics Section was formed in January 1982 when the Solid Mechanics Section was split into two sections—Structural Mechanics and Pressure Vessel Technology. The Structural Mechanics Section included two former Solid Mechanics groups—the Engineering Mechanics Group, which was formed in 1974, and the Nuclear Standards Management Center, which was established in 1978. J. M. Corum had the dual responsibility for both these groups, and he was named to head the Structural Mechanics Section.

This historical overview covers the mechanics and related materials activities of the Engineering Mechanics Group and the Structural Mechanics Section. From 1975 to 1992, research efforts and expertise broadened as successful marketing efforts yielded funding from a variety of sponsors. Throughout these years, the DOE-sponsored High-Temperature Structural Design (HTSD) Program and nuclear-safety-related NRC projects involving piping and nozzles served as the cornerstone for the section. Building upon these efforts, the section undertook major initiatives, both nuclear and nonnuclear, for several DOE offices and various DOD departments between 1985 and 1992. This

*In November 1986, the HFIR was shut down because of concerns about RPV embrittlement. Following 3 years of comprehensive reviews and testing, the reactor was restarted in April 1989; however, an inadvertent shutdown the following month led to additional reviews. In January 1990, HFIR operations were resumed with substantial procedural changes and a reduction in maximum power level from 100 to 85 MW(t). Due to tighter controls and operating restraint, the greatest possible maximum-power-level operating time in a 21-d cycle of perfect operation is reduced to 78%, down from the previous record of 98%.

expansion in scope ranged from nuclear safety standards, design criteria, and advanced reactor concepts to composite materials technology for the armed forces and aerospace defense systems. Shifts in research emphasis reflected responses to changing challenging technological needs within the nation. Transitions were gradual, however. The NRC safety-related work was the forerunner of several programs and is first given as background.

In 1974 when the AEC was split into the NRC and ERDA, the NRC took control of the former AEC safety-related programs including the ORNL Nozzles Program and the ORNL Piping Program. NRC then combined the two programs into the single ORNL Nozzle and Piping Program in 1975. S. E. Moore was assigned to manage the combined program.

Between 1975 and 1979 the work of the Nozzle and Piping Program was largely that of phasing out the remnants of the two earlier programs and documenting results. By 1981 this program and its two predecessors had yielded 235 technical reports and papers. Essentially all of the program results have been incorporated into the *American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code*, Sect. III for nuclear components, either directly as Code revisions or indirectly in support of the Code rules. The Nozzle and Piping Program was officially phased out in 1981.

Beginning in 1981 and continuing through 1986, the section worked for the NRC Office of Nuclear Reactor Regulations in support of design documentation audits for nuclear power plants seeking operating licenses. Section staff examined design documentation on piping, pumps, valves, and supports that was submitted to the NRC for 13 nuclear power plants. A critique was written on the design documentation practices of the utilities in trying to satisfy the needs of both the NRC and the *ASME Code*. Published in 1987 and presented to a special *ASME Code* panel in 1988, the NRC report recommends major changes in the Code that are currently under consideration.

A project to provide the NRC Office of Nuclear Regulatory Research with technical assistance in areas related to the *ASME B&PV Code* was initiated in 1982 under the guidance of G. T. Yahr and is still continuing today. S. E. Moore, with subcontracted assistance from E. C. Rodabaugh, has provided considerable assistance in the area of piping and nozzles. R. C. Gwaltney conducted a joint study with Idaho National Engineering Laboratory (INEL), operated by EG&G, to compare the Code Sect. III with Sect. XI rules for fatigue crack growth. Much of the work has been directed at improving the current approach to seismic design of piping systems. Yahr developed guidance for the design and preload of bolted joints to alleviate a persistent problem with stress corrosion cracking.

Sponsored first by AEC and later by DOE-NE, the HTSD methods development program, which had been established at ORNL in 1969 to support the national LMFBR Program, had by 1975 become one of the Reactor Division's largest activities. The program was, according to the DOE Oak Ridge Operations Office, the single most important ORNL R&D task. From the middle to late 1970s, the task involved most of the present-day Structural Mechanics Section, as well as a substantial supporting effort from the Metals and Ceramics (M&C) Division. W. L. Greenstreet guided the program through its formative early years. H. C. McCurdy then became manager until 1976 when J. M. Corum, with the assistance of C. E. Pugh, took over the program management. The program continued, although at a decreasing level in recent years, until late in 1991—a total of nearly 22 years! For the last 3 years, the effort was jointly supported by the Japan Atomic Power Company and DOE, and J. J. Blass was in charge.

The importance of this program was based on the fact that LMR components presented unique structural design requirements. In the late 1960s, it was recognized that the low-temperature structural design methodology developed and used for LWRs would not be adequate for LMRs. ORNL's task was to develop an HTSD methodology that explicitly accounts for the effects of nonlinear material deformation and time-dependent damage mechanisms and failure modes—something that had never before been done.

In addition to the in-house work, ORNL was given a management role with respect to other participants in the HTSD technology area—most notably Westinghouse, Rockwell International, General Electric, and the Hanford Engineering Development Laboratory (HEDL). ORNL also coordinated international exchange meetings, workshops, and collaborative efforts in the HTSD methods area primarily with the United Kingdom, France, Germany, and Japan throughout the 1970s and 1980s.

The upshot of this major effort was the successful development and experimental validation of a methodology that has been accepted and used worldwide. Both inelastic design analysis methods and simplified methods were established, and these are specified in a 1986 DOE design guideline.* Likewise, criteria for guarding against structural failures were developed and are given in the *ASME B&PV Code* Case N-47 for design of high-temperature nuclear components. Most of the basic elements of this multifaceted HTSD methodology have been experimentally validated, and the methodology has been successfully used in the design of the Fast Flux Test Facility at Hanford, Washington, and the Clinch River Breeder Reactor Plant (CRBRP), construction of which began, but was never completed, in Oak Ridge in the latter part of the 1970s. The division staff played a role in the CRBRP licensing hearings before NRC in 1982, and much of the direction of the development program in recent years was shaped to answer some of the NRC concerns with the new technology.

*NE Standard F9-5T, *Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components*, September 1986.

The primary areas addressed by division staff members and some of the individuals involved are listed below:

- material testing and deformation and failure modeling: R. L. Battiste, J. J. Blass, J. M. Corum, J. R. Ellis, W. L. Greenstreet, R. L. Huddleston, K. C. Liu, C. E. Pugh, D. N. Robinson, M. B. Ruggles, and W. K. Sartory

- inelastic structural analysis methods and computer codes: J. A. Clinard, Y. L. Lin, and W. K. Sartory

- confirmatory structural testing: R. L. Battiste, J. M. Corum, A. G. Grindell, W. J. McAfee, M. Richardson, and H. C. Young

- weldment design considerations: T. J. Delph, W. R. Hendrich, W. J. McAfee, and D. G. O'Connor

- simplified methods: R. C. Gwaltney, G. T. Yahr, and W. K. Sartory.

- standards development: J. J. Blass and J. M. Corum.

In addition to design guidelines and design criteria, a third ingredient needed by the high-temperature nuclear component structural designer/analyst was a body of approved materials data. The 4-volume, 16-book DOE *Nuclear Systems Materials Handbook (NSMH)* provides those data for LMRs and other high-temperature reactor systems. In 1982, development and management of the NSMH was transferred from HEDL to ORNL and assigned to ETD. M. F. Marchbanks moved from HEDL to ORNL and became the ETD manager of the effort. Over the years, additional materials data systems—handbooks, as well as computer data bases in some cases—have been developed under Marchbank's direction.

The section also participated in LMFBR seismic studies for DOE. Between 1983 to mid-1985,

R. C. Gwaltney coordinated the development of guidelines for the seismic ground motion definition for the Eastern United States under the LMFBR Program. During the seismic study, he also coordinated and monitored a subcontract with Agbabian Associates to publish a report on its 12 years of seismic work under the LMFBR Program.

Over the years, the HTSD Program led to several related activities, some for sponsors other than DOE. Parallel efforts for NASA and EPRI began in the mid-1980s.

Beginning in 1984, ORNL carried out an experimental effort for NASA-Lewis Research Center designed to measure multiaxial flow surfaces using tubular specimens of type 316 stainless steel at 1200°F. This work supported NASA's efforts to improve the design methodology for the hot sections of aircraft engines. Flow surface determinations were made after certain torsional preloadings. The flow surfaces formed the basis of a viscoplastic constitutive theory that reduced assumptions concerning the multiaxial stress dependence. Principal investigators were J. A. Clinard and R. L. Battiste.

In 1986 ORNL was requested to participate in collaborative LMR development studies conducted by EPRI in the United States, the Central Research Institute of Electric Power Industry (CRIEPI) in Japan, and the Nuclear Electric plc (NE, formerly the Central Electricity Generating Board) in the United Kingdom. ORNL activities were to provide a comparative assessment of candidate constitutive theories for use in inelastic design analyses of high-temperature components of advanced LMR plants. J. J. Blass led the ORNL effort, with participation by R. L. Battiste, S. J. Chang, Y. L. Lin, and W. K. Sartory.

In 1988 ORNL participation in the joint studies shifted to identification of a high-temperature flaw assessment procedure for reactor components. As a result of a 2-year collaboration, an interim high-temperature flaw assessment guide was produced under the overall coordination of ORNL. The procedure addressed pre-existing defects in high-temperature reactor components subject to creep-fatigue conditions. M. B. Ruggles led this phase of the joint study with experimental support provided by R. L. Battiste. In 1991 Ruggles began leading a new 2-year EPRI/CRIEPI/NE collaborative study on inelastic behavior and creep-fatigue criteria for modified 9 Cr-1 Mo steel at elevated temperatures.

In the late 1980s the division provided support to DOE for the MHTGR Program by playing a lead role in developing an *ASME B&PV Code* case for very high temperature design of components for process heat and direct-cycle reactors, where temperatures to 1800°F are envisioned. J. M. Corum, as a member of the Code Subgroup on Elevated-Temperature Design, which has development responsibility for elevated-temperature rules, participated in an ad hoc Code committee for this effort. J. J. Blass and S. J. Chang helped by developing constitutive equations for nickel-base alloy 617, which is the primary material of interest. K. Hada, who was on a 1-year assignment to ETD from the Japan Atomic Energy Research Institute,* helped with various background studies. This effort resulted in a proposed new Code case currently being reviewed by higher Code bodies.

In December 1985 INEL asked R. C. Gwaltney for ORNL assistance in the EG&G–TVA Weld Evaluation project at the TVA Watts Bar Nuclear Reactor Plant. ORNL participated in both weld inspection and reanalysis of the existing welds until early 1987. F. C. Zapp, J. J. Blass, and S. J. Chang worked at the Knoxville TVA Office; G. T. Yahr, Gwaltney, C. R. Luttrell, S. E. Moore, and D. G. O'Connor worked at the Watts Bar Plant. W. C. Cooper helped coordinate the activities of the ORNL personnel at the TVA sites and ORNL.

*Hada was one of four Japanese assignees to ETD in the HTSD technology area. The first was K. Iwata from the PNC Oarai Engineering Center in 1982. More recently, Y. Takahashi and T. Ogata from CRIEPI have spent a year or more each at ORNL.

As part of a DOE Defense project in 1987, ORNL was asked by Savannah River Laboratory to develop Preliminary Acceptance Criteria for assessing the performance of the emergency core cooling systems for HWRs in response to a LOCA. The acceptance criteria that were developed are the equivalent of the 1973 acceptance criteria for LWRs incorporated within 10 CFR 50–46. The work spanned a period from 1987 to 1990 with T. E. Cole, R. C. Gwaltney, R. P. Wichner, C. R. Luttrell, and M. F. Marchbanks working on the project.

As noted earlier, the Structural Mechanics Section's scope broadened to encompass a variety of R&D activities for DOD, involving the U.S. Navy, U.S. Air Force, and U.S. Army from 1986 to 1989.

For the Systems and Equipment Maintenance Monitoring for Surface Ships (SEMMSS) Program sponsored by the Naval Sea Systems Command, D. G. O'Connor and W. F. Swinson (then a summer participant from Auburn University) analyzed failed parts of fire pump impellers to determine the root cause of failure and to recommend appropriate inspection procedures.

Also, as part of the SEMMSS Program, J. A. Clinard led engineering analysis and software development efforts that resulted in computer-aided performance trending analysis software encompassing 35 distinct ship systems. This software was subsequently incorporated into the Navy's predictive maintenance program for production use. J. C. Moyers (Engineering Analysis Section) and L. Jung (Thermal Systems Technology Section) were responsible for the engineering analysis of the equipment systems in preparation for the software development task, which was largely performed by J. J. Robinson.

In 1987, R. L. Battiste and others installed more than a thousand strain gages on two submarine models in an extremely high quality fashion and on a very tight schedule. This work was in support of the SSN-21 submarine class project and was sponsored by the David Taylor Naval Ship R&D Center. These models were then shipped back to the R&D Center for design verification testing.

Also from 1986 to 1991 ETD began a new initiative in composite materials and structures technology. A major emphasis was on carbon-carbon composites (C-C). The objective was to use Oak Ridge's three-plant base of expertise to attract interesting, nationally important programs and to broaden ORNL's and Oak Ridge's base of funding. R. L. Huddleston provided overall leadership for the carbon-carbon initiative, including marketing and program management.

With resources from the Lab Seed and Directors' R&D funds, a new facility was brought to an interim state of completion in 1989 by D. G. O'Connor. The new facility provided ETD and ORNL with a unique capability to test materials at extreme temperatures (~4000°F ) in air, which is unique both within the United States and worldwide. This facility is key to developing advanced surface-protected C-C materials technology to meet aerospace and other needs.

The marketing effort associated with and carried out in parallel with the Seed and Directors' R&D projects was also successful in attracting new funding for interesting state-of-the-art projects primarily from the U.S. Air Force (USAF), Wright-Patterson Air Force Base. A new multi-year "Carbon-Carbon Applications Project" was initially directed at technology assessment and, subsequently, at design and development of a C-C structural material technology for a new unmanned aerospace vehicle (UAV) to fly at hypersonic speeds up to Mach 16 to 20 within the atmosphere. The multidisciplinary project was managed by R. L. Huddleston with matrix participation by W. K. Sartory, ETD; R. A. Lowden of M&C; C. W. Haaland of Engineering Physics and Mathematics; G. E. Wrenn, A. J. Caputo, and C. D. Reynolds of Y-12 Development; and C. Holcombe of Engineering. The combined team effort from early 1986 to late 1990 led to a new coated C-C material system concept with the potential to meet USAF needs and one that was picked up as baseline by the USAF prime contractors.

In mid-1988 the USAF gave ORNL a second important program assignment—to develop a strategic investment plan for nonmetallic materials and structures for advanced aircraft and aerospace vehicle airframes. Project elements included determining future USAF missions and systems needs at the major command and Pentagon level (encompassing fighters, bombers, tankers, transports, and UAVs); establishing data bases for materials currently certified for airframe design and new and innovative materials; conducting trade studies to quantify new materials' performance payoffs in advanced vehicles; quantifying mission payoffs; defining ongoing R&D efforts and gaps; and finally making recommendations for technology investment. Huddleston provided the overall methodology for the study and overall program management with W. L. Greenstreet and, subsequently, W. F. Jones managing the project. Support was provided by Y-12 as well as by the M&C Division and subcontractors such as SAIC-Washington and LTV Corporation. This project was successfully completed in late 1990.

Other noteworthy projects conducted under the new C-C initiative included the "Graded Hybrid Coatings Project," the "C-C Rapid Densification Project," and the "C-C Brazing Project." D. G. O'Connor led the advanced rapid densification technology development project with the technology jointly developed by ORNL and Textron (under subcontract) for the USAF during the period 1988 to 1990. This effort was very successful in demonstrating that a new process being pioneered by Textron has the potential to greatly reduce processing time and cost for C-C and thus should have a major payoff for DOD.

Overall the C-C initiative was very successful for ETD, ORNL, and Oak Ridge: (1) it provided ORNL with a unique testing capability that can help gain participation in major DOE, DOD, and NASA programs because it can satisfy technology needs of nationally important programs; (2) it has demonstrated that ORNL can attract and execute important multidisciplinary programs in the composites arena; and (3) it created credibility in the C-C and composites community within DOD and NASA that can assist ETD and ORNL in obtaining future funding in this arena.

Another area of involvement with composite materials and structures has been the use of advanced materials to lighten military structures. G. T. Yahr, C. R. Luttrell, R. C. Gwaltney, with support from J. A. Mayhall of Engineering, R. E. Norris of the Applied Technology Division, and D. G. O'Connor, provided support to the Army Materials Technology Laboratory (MTL) in the application of composites to lightening of howitzers. They conceived a new trail attachment scheme for the 155-mm M-198 howitzer and developed a bottom carriage design using boron carbide particulate-reinforced aluminum that reduced the weight by 55%. MTL was also assisted in its activities to lighten the 105-mm M-102 howitzer.

A second activity in lightweight structures addressed airdrop platforms in a project conducted for the Army Natick Research Development and Engineering Center; W. R. Hendrich developed a lighter E-glass/epoxy drop platform design to replace the current aluminum platform.

As a final DOD-sponsored activity, the section was involved in the Advanced Shield Phenomenology Program, which ETD managed for the SDI. The goal was to provide a low-weight survivable shield design for orbiting spacecraft. Hendrich used a hydrocode* on the CRAY computer to predict the response of spaced-array shields to hypervelocity projectiles. His analyses

*A hydrocode employs an analysis scheme based on conservation of energy, momentum, and mass of a volume element of material. Such codes are particularly adept at treating hypervelocity impact problems, where materials become fluid.

helped guide the design of shields that were subsequently tested at the Arnold Engineering Development Center in Tullahoma by J. E. Smith and D. G. Thomas. These shields had the same shielding performance as solid aluminum shields that were ten times heavier.

To further enhance ETD's and ORNL's role and capabilities in composite materials and structures, ETD in 1989 worked with the University of Tennessee's (UT's) Department of Engineering Science and Mechanics to obtain an internationally known composites expert, jointly appointed under the UT/ORNL Distinguished Scientist Program. Dr. Y. J. Weitsman, who shares his time between the Structural Mechanics Section and the University, has focused his ORNL effort on development of a multifracture model of continuous fiber ceramic composites. This promising development will aid both material developers and designers.

As a result of DOE establishing its Office of New Production Reactors, ORNL received funding in 1989 for work in a variety of development areas to support the design and construction of an NPR. ETD's major involvement was and is in the area of Materials and Structures R&D to support both the HWR and the MHTGR reactor concepts for tritium production.

The HWR Materials and Systems Integrity Task, managed by J. A. Clinard, is conducted in the ETD and M&C Divisions. The multifaceted task includes activities of materials selection, aluminum corrosion, stainless steel corrosion, irradiation effects, nondestructive evaluation methods, component fabrication technology, design methods, and fracture margin assessment methods. A preliminary materials properties handbook was developed by M. F. Marchbanks and D. G. O'Connor. The task addresses all identified materials data and system integrity needs of the primary boundary components including the reactor vessel, piping, pumps, heat exchanger, etc. Much effort to understand and quantify possible corrosion and irradiation degradations is being expended to extend the design life to 60 years.

One significant subtask of the HWR Materials and Systems Integrity Task involves a series of impact tests of full-size piping components. Building on efforts by R. C. Gwaltney to define leak-before-break methods* for the HWR primary piping, A. B. Poole designed the subject set of experiments to further discredit the double-ended break as a credible failure mode for the HWR primary piping. A sophisticated test fixture and experimental apparatus were constructed by Poole and R. L. Battiste.

In support of the New Production MHTGR, which is to generate electric power in addition to producing tritium, section staff are working to establish structural analysis methods and design criteria for graphite core support components and metal heat-transport system components. G. T. Yahr, W. F. Swinson, R. L. Battiste, and M. F. Marchbanks are preparing to conduct a series of tests of tubular specimens of core support graphite under combinations of axial load and internal pressure to establish the form of the multiaxial strength criterion. J. J. Blass and R. L. Battiste are preparing to conduct tests and analysis of tubular specimens containing prototypic welds, like those joining austenitic Alloy 800H and ferritic $2\frac{1}{4}$ Cr-1 Mo steel tubing in the steam generator, to establish design criteria and life-assessment procedures.

Throughout the years, ORNL divisions have supported one anothers' efforts with various multidisciplinary projects. Thus, Structural Mechanics also supported development of the ANS, AVLIS Program, and magnetic fusion energy projects for DOE.

R. C. Gwaltney took the early lead for identifying R&D needs in the materials and structures area for the replacement for the HFIR, that is, the HFIR-II

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*Leak-before-break is a safety analysis concept, accepted for limited use by NRC, that proves that a component will leak before it breaks.

and CNR, now called the ANS. G. T. Yahr, current Task Leader for the ANS "Materials Data, Structural Tests, and Analysis R&D," suggested that the primary pressure boundary should be close to the reactor core so that numerous guide tubes and beam tubes would not penetrate the primary pressure boundary. He and M. F. Marchbanks evaluated candidate materials for use as the primary pressure boundary for the ANS and selected 6061-T6 aluminum. Yahr then prepared a request, and a suggested response, to the ASME Code Committee asking for rules using aluminum alloy 6061-T6 as a material for Class 1 nuclear components in Sect. III.

Assessing the structural performance of the ANS reactor fuel plates is also the section's responsibility. Past experience has shown that fuel-plate failures can occur when the coolant flow causes the closely spaced plates to deflect and touch, resulting in burnouts. Because the ANS has a very high power density that requires a higher coolant flow velocity than previous reactors, potential is higher for plate instability problems. W. K. Sartory developed an improved instability analysis of the involute fuel plates by coupling curved shell equations for the involute fuel plates to two-dimensional hydraulic channel flow equations that include fluid friction. W. F. Swinson and C. R. Luttrell then verified the accuracy of the analytical method by testing epoxy involute plates.

W. R. Hendrich evaluated the potential for flow-induced vibrations in the ANS control rods and is responsible for the Control Element Test Facility, which will be used to evaluate the performance of the control rods under realistic flow conditions.

ETD supported the AVLIS Program by conducting analytical and experimental studies to ensure structural integrity and alignment of the graphite collector structure. In 1984, Y. L. Lin and R. C. Gwaltney analyzed the graphite vapor collector structure both for buckling and thermal loading. D. G. O'Connor and W. R. Hendrich tested parts of the structure to determine their strength. G. T. Yahr and O'Connor provided structural design criteria and design data for AVLIS graphite components.

As a sister research division to Fusion Energy, ETD has served in a support capacity to several DOE magnetic fusion energy projects. Through 1987, J. A. Clinard supported the International Large-Coil Test Program at ORNL by performing the large, complex structural analyses of the test facility and six different superconducting magnets (coils) necessary to confirm magnet integrity for test conditions specified by the project. These state-of-the-art analyses were performed on a CRAY computer. The results were featured by Cray Research, Inc., in *Cray Channels*, a promotional publication.

## 5.6 PRESSURE VESSEL TECHNOLOGY

The Pressure Vessel Technology (PVT) Section came into being in 1982, when the Solid Mechanics Section was divided into the Structural Mechanics and PVT Sections. G. D. Whitman was appointed as Section Head of PVT. At this time the PVT Section consisted primarily of the Heavy-Section Steel Technology (HSST) and the Prestressed Concrete Pressure Vessel (PCPV) Programs, both of which came into existence in the mid-1960s.

Since 1975, the HSST Program has been sponsored by NRC and the PCPV Programs by both the NRC and DOE. In 1975 G. D. Whitman headed the Solid Mechanics Section and also managed the HSST Program, while J. P. Callahan managed the PCPV Program. Upon Callahan's departure from ORNL in 1979, D. J. Naus took over the PCPV Program. When the Solid Mechanics Section was divided in 1982, C. E. Pugh became manager of the HSST Program. G. D. Whitman was elected a Union Carbide Corporation Corporate Fellow in 1983 and later retired in 1986, at which time C. E. Pugh assumed leadership of the PVT Section. Shortly thereafter, in 1986, W. R. Corwin became manager of the HSST Program.

Major objectives of the HSST Program have been (1) development of methodology for predicting flaw behavior in RPVs, (2) experimental investigation of validity of fracture-mechanics-oriented predictive methodologies, (3) irradiation of vessel materials and subsequent testing to establish an irradiation effects data base, and (4) development and application of methodology for evaluating integrity of RPVs and their structural supports.

J. G. Merkle has been primarily responsible for the development and evaluation of basic fracture-mechanics analytical methodologies, with contributions from R. H. Bryan, J. W. Bryson, J. S. Parrott, W. E. Pennell, M. N. Raftenberg, D. K. M. Shum, and G. C. Smith. Task leaders for large-scale confirmatory experiments were R. H. Bryan, who led testing of 18,000-lb flawed vessels with pressure and pressure plus thermal-shock loading; R. D. Cheverton, who directed testing of 10,000-lb flawed cylinders with thermal-shock loading; and D. J. Naus, who led testing of 25,000-lb flawed plate-type tensile specimens. R. W. McCulloch was the lead engineer for the design of the pressurized-thermal-shock (PTS) facility, and G. C. Robinson was responsible for the detailed mechanical, thermal, and hydraulic design of this and other experiments. J. E. Smith was responsible for developing instrumentation, and S. E. Bolt and P. P. Holz were responsible for preparation of the experimental facilities. Task leaders for smaller scale experiments were R. D. Cheverton and W. J. McAfee for efforts in cladding effects and warm prestressing and T. J. Theiss for activities involving crack-arrest and shallow-flaw fracture toughness.

In the process of developing and applying fracture-mechanics analytical methodologies, numerous computer codes have been written. Some of the more widely used of these codes are ORMGEN and ORNOZL, two- and three-dimensional mesh generators; ORVIRT, a fracture-related postprocessor for the well-known structures code ADINA; a modified version of ADINA for dynamic analysis of cracks; and OCA-P, a deterministic and probabilistic fracture-mechanics code for evaluating the integrity of RPVs.

The HSST Program continues to address the licensing needs of the NRC with concentrated efforts in the areas of constraint effects on fracture toughness, cladding effects on the potential for propagation of surface and subclad flaws, suppression of flaw propagation by warm prestressing (crack-tip conditioning), dynamic effects on crack propagation in reactor vessels, and updating of the OCA-P code. PVT personnel involved in these ongoing efforts are W. E. Pennell (HSST Program Manager), J. W. Bryson, J. G. Merkle, G. C. Robinson, D. K. M. Shum, and T. J. Theiss.

The HSST Program has relied upon input from many personnel outside ETD—particularly the M&C and Computing and Telecommunications Divisions—and outside ORNL—universities, industry, and other national laboratories in the United States and elsewhere. The total HSST effort has contributed to the development and updating of national codes, standards, and regulatory guides that are helping to evaluate and regulate the safe operation of nuclear RPVs and their structural supports.

The PVT Section's second major effort, the PCPV Program, underwent parallel evolution and development into a comprehensive concrete R&D program in support of several advanced energy systems. Although the program has been supported by many sponsors over its 25 plus years of existence, DOE and NRC have provided primary support. DOE-sponsored concrete program activities have addressed development of PCRVs for gas-cooled reactor concepts and coal gasification facilities and support for development of the breeder reactor.

Under the PCPV Program, analytical and experimental studies were conducted to support the HTGR and the GCFR concepts. HTGR activities involved four basic but interrelated tasks: (1) technology assessment, which focused on containment concepts and practices, concrete embedment instrumentation, prestressing steel corrosion inhibitors, optimized PCRV for HTGR steam cycle plant, and steel reinforcement and prestressing systems; (2) analysis methods development, which included inelasticity and failure analyses and support for model tests; (3) material studies, which encompassed concrete creep behavior, elevated temperature behavior, development of high-strength concrete mix designs, and multiaxial behavior; and (4) model testing, which focused on thermal cylinder, head failure studies, and moisture migration. Activities under the HTGR Program were terminated in 1985 when the program switched to a modular design that used a steel RPV. Under the GCFR Program, three concrete model tests were conducted between approximately 1975 and 1981 to verify design of the closures for the steam generator and central core cavities for the PCRV of a 300-MW(e) plant.

Analytical studies for these two programs were conducted by W. G. Dodge, D. N. Fanning, J. R. Dougan, and M. F. Raftenberg. The experimental work was performed by D. J. Naus, C. B. Oland, and G. C. Robinson, Jr.

From 1976 to 1977 under W. L. Greenstreet's leadership, studies were conducted on the use of PCRVs for commercial-size coal gasification systems. Problem areas were identified, and a test program was defined for concept verification and performance examination and demonstration. C. B. Oland developed conceptual vessel designs for two gasifier systems, the HYGAS and Synthane processes. Results of these studies indicated that the use of PCRVs was both technically and economically feasible in applications where large, heavy-walled steel vessels were formerly used.

In support of the CRBRP design, D. J. Naus and C. B. Oland conducted an elevated-temperature test program between 1978 and 1981. This study evaluated the variations in mechanical properties, such as compressive stress and strain behavior, shear strength, concrete-rebar bond, and creep and thermal properties, such as coefficients of thermal expansion, diffusivity, and conductivity of a limestone aggregate concrete and a lightweight insulating concrete exposed to high temperatures. Because of the temperature exposure of interest (up to 1150°F ), the program required the development of specialized test methods and instrumentation systems. C. B. Oland and G. C. Robinson conducted this latter effort.

In 1981, funding support for the concrete program began changing from DOE to NRC. J. R. Dougan conducted the first of the NRC-sponsored activities; it involved an analysis of the in-service inspection requirements for greased tendons of posttensioned concrete containments of LWR facilities. This activity was followed in 1982 by J. R. Dougan's technical review of existing guidelines for leak-rate testing of LWR containments.

In 1984 the Naval Reactors Branch of DOE sponsored a new program, the Thermal Shock Studies Program, which was added to the section. R. D. Cheverton assumed leadership of the Thermal-Shock Studies Program, which in many respects is very similar to the HSST Program; it focuses on the effects of cladding and warm prestressing on the behavior of surface and subclad flaws in RPVs during PTS loading conditions. R. D. Cheverton also managed the first phase of the program in which large steel test cylinders (~10,000 lb), clad on the inner surface and containing multiple flaws, were subjected to severe thermal-shock loading, and a new laboratory-size clad specimen (Jo-Block) was developed and tested. W. J. McAfee directed the second phase of the program that continued the Jo-Block specimen development and testing, including testing of irradiated specimens, and involved the investigation of warm prestress effects with the testing of clad and unclad beams in three-point bending. G. C. Robinson performed the detailed design of testing machine fixtures and the test specimens.

During 1986 D. J. Naus conducted a study for the Nuclear Plant Aging Research Program to evaluate concrete component aging and its significance relative to life extension of nuclear power plants. The results of this study were used to help formulate a multiyear Structural Aging (SAG) Program, which was initiated in 1988 and is presently ongoing. The overall objective of this new program is to provide NRC with a report that identifies potential structural safety issues and acceptance criteria for use in nuclear power plant evaluations for continued service. The initial focus of the SAG Program is on concrete and concrete-related materials that comprise safety-related structures in LWR facilities. The program is organized into a management task and three technical task areas: material properties data base, structural component assessment/repair technology, and development of a quantitative methodology for facilitating a continued-service determination. An important product of this program thus far has been the establishment, by C. B. Oland, of the Structural Materials Information Center. D. J. Naus, Program Manager, and C. B. Oland have received commendations from the NRC for their management of, and technical contributions to, the SAG Program, which requires the coordination of numerous subcontractors in the United States and abroad.

In 1989, the irradiations portion of the HSST Program was established as a separate program, the Heavy-Section Steel Irradiations Program. W. R. Corwin was appointed to manage this new program, and W. E. Pennell assumed leadership of the HSST Program. At about this same time, C. E. Pugh became Director of NRC Programs at ORNL, R. D. Cheverton became the PVT Section Head, and W. J. McAfee became manager of the Thermal Shock Studies Program.

Management of the HSSI Program was then transferred to the M&C Division, leaving the PVT Section with three major programs: Structural Aging, formerly the "Concrete" Program; HSST, without the irradiations portion; and Thermal Shock Studies. Activities for these programs are ongoing.

In addition to these three major efforts, PVT personnel have participated in many other related research activities. These activities include (1) the Integrated Pressurized-Thermal Shock Program, which was sponsored by NRC and managed by ORNL Engineering Physics/I&C Divisions with participation by R. D. Cheverton and which helped to establish a PTS evaluation probabilistic methodology and the NRC PTS rule used by NRC; (2) evaluation of structural support integrity for LWR pressure vessels considering radiation embrittlement with R. D. Cheverton, W. E. Pennell, and G. C. Robinson as major contributors; (3) reevaluation of pressure vessel integrity for the HFIR, which involved efforts by R. D. Cheverton and J. G. Merkle; (4) evaluation of Savannah River NPR pressure vessel integrity conducted by J. G. Merkle; (5) evaluation of appropriateness of NRC PTS Rule for two reactors in Belgium performed by R. D. Cheverton and J. G. Merkle and sponsored by a Belgium utility; (6) evaluation of reactor vessel integrity for the Yankee Rowe Nuclear Plant with contributions from R. D. Cheverton and J. G. Merkle under NRC sponsorship; (7) evaluation of structural aspects of an advanced Canadian reactor design involving G. C. Robinson and W. J. McAfee; (8) consultation with NRC-NRR regarding pressure vessel integrity with participation of R. D. Cheverton, J. G. Merkle, and W. E. Pennell; (9) evaluation of nuclear-plant snubbers conducted by J. H. Butler and sponsored by DOE; (10) development of a method for creating specific flaws in experimental steel shipping casks by G. C. Robinson under DOE sponsorship; and (11) development and application of two-phase flow instrumentation for the German–Japanese, NRC 2D-3D Refill/Reflood Program, which involved J. E. Smith. The program's purpose was to examine emergency-core-cooling-water behavior of a PWR after a LOCA.

## 5.7 SPACE AND DEFENSE TECHNOLOGY PROGRAM

From a new sponsor evaluation project requested by H. E. Trammell, S. R. McNeany proposed the development of optic programs with the U.S. Army Strategic Defense Command of Huntsville, Alabama. Initial work involved programs managed by C. Martin and L. Atha. At Oak Ridge, work began with the turning of optical mirrors at the Y-12 Plant, research on various materials at ORNL, and the initiation of the Optical Characterization Laboratory (OCL) at the K-25 Site.

In the fall of 1987, Lt. Col. B. Brown of the Strategic Defense Initiative Organization (SDIO) contacted D. E. Bartine, Director of the SDI Program, to discuss additional optics R&D work that would be funded directly from Washington. The driver behind this effort was a desire on the part of Lt. General Abrahamson, Director of SDIO, to address manufacturing issues of generic but critical optics components before the design is fixed. Based on the General's experience with the NASA Shuttle and DOD's F-16 Program, he encouraged the development of a Manufacturing Operations Development and Integration Laboratory (MODIL). L. Fehenbacher, G. Stottlemyer, and Lt. Col. B. Brown worked with several potential integration contractors but selected Energy Systems as the most likely candidate. That selection was primarily based on the Y-12 Plant's experience with the machining of beryllium—a lightweight, but strong, material with advantages for space applications.

Initial funding arrived in November 1987 to begin preparing a plan. W. R. Martin assumed the leadership role to develop the strategy and formal plan for MODIL. On February 10, 1988, that plan was presented to General Fox, Deputy Director of SDIO, and he concurred. The result was increased funding for that current fiscal year to begin execution of the plan. The Space and Defense Technology Program was created with W. R. Martin as Program Manager and R. Steele as Technical Manager.

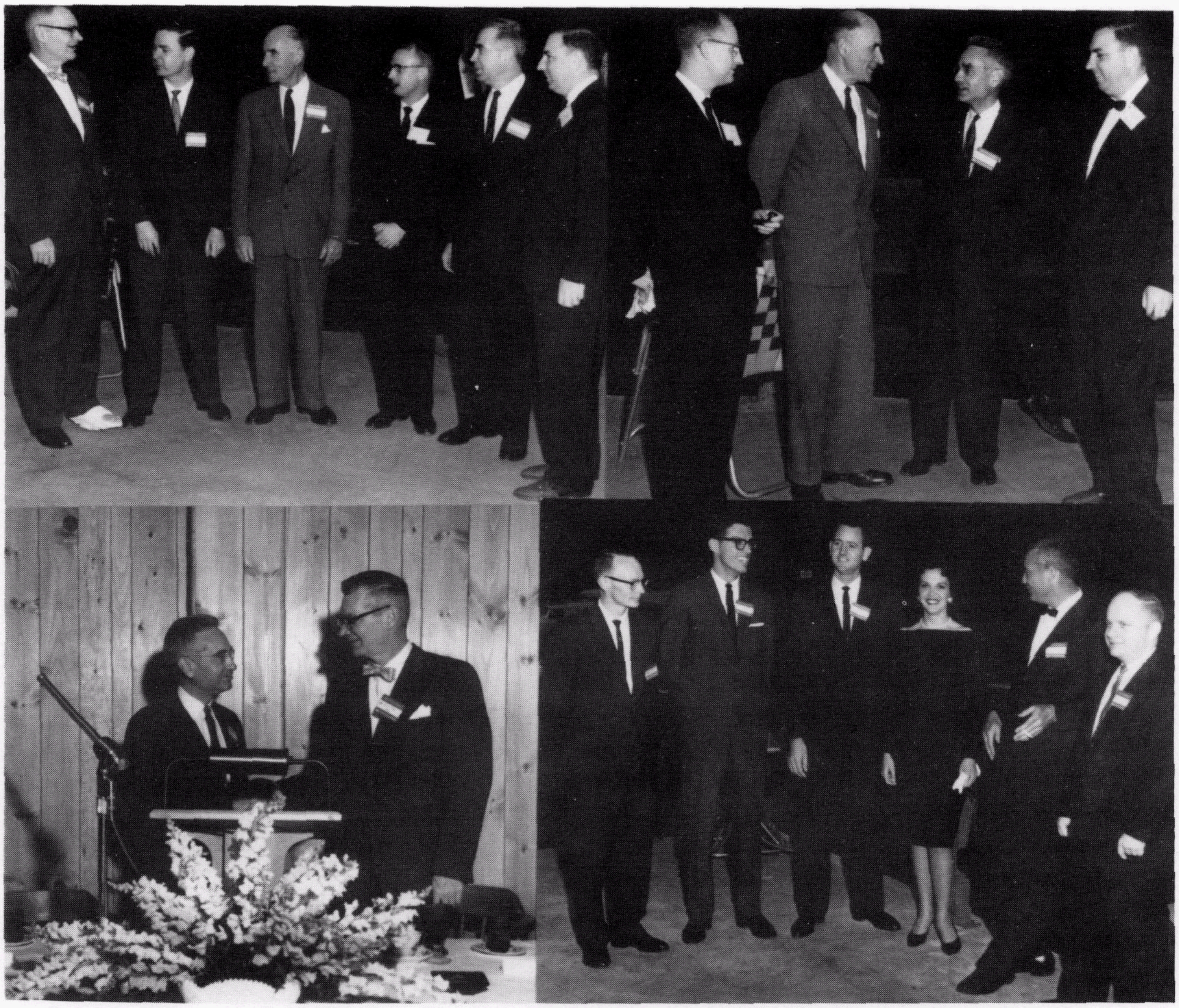
Industrial briefings were arranged and chaired by W. R. Martin in March and June of 1988. An approach that included briefings and workshops was designed to attract industrial participation. A host of researchers from Y-12 Development Division and ORNL began extensive traveling across the country to determine the real issues in the manufacturing of optics. The goal was to make high-quality optics more quickly and efficiently and, within 10 years, to reduce the cost of optics by an order of magnitude.

In February 1989, the Optics MODIL began to award contracts to other companies and universities. Research began on manufacturing technology; J. A. Wheeler became Operations Manager, and P. Steger was Technical Manager. By 1990, the Producibility and Validation Test Bed was established at ORNL. One of the world's most precise turning machines was installed.

In a parallel effort, the Advanced Optics Materials Development Program was being directed by W. B. Snyder, with M. A. Akerman as his Technical Manager. This program had brought the OCL to the level that now involved testing a large number of DOD contractor samples before and after underground testing. Work on the materials side had concentrated on diamond films and developing boron carbide as a suitable baffle material.

By 1991, the efforts of both programs were supported by significant funding and involved more than 50 personnel at Energy Systems. Over 400 industry, university, and federal laboratory personnel have attended one or more of the ten industrial briefings. The optics program has national visibility and attention.


***Reactor Division dinners were held into the early 1960s. Pictured at the 1963 dinner are (clockwise from top left) Dave Ghormley, Joel Witt, John Merkle, and Jack Smith; Bill McDonald, Clyde Claiborne, and H. G. MacPherson; unidentified, McDonald, Jim Lane, and Walt Jordan; Lou Parsley, Mike Bender, and Sam Beall.***



*The First Southeastern Conference on Theoretical and Applied Mechanics was held in Gatlinburg under the sponsorship of ORNL. This biennial, regional conference was established in 1962 to help strengthen applied mechanics in the southeastern states. The conference had its beginnings in the Reactor Division's Applied Mechanics group, and it had full support and help from the Laboratory as well as support and backing of faculty members from southeastern universities and industry representatives. Bill Greenstreet was chairman of the first conference, and Herb Hoffman was program committee chairman. To date, 16 conferences have been held; 15 have been hosted by 12 universities. Pictured at the first conference are (clockwise from top left) Dick Lyon, Grover Rodgers of Florida State University, J. P. Den Hartog (well-known mechanician, MIT professor, and first keynote speaker), Bill Greenstreet, Ross Evan-Iwanowski (Syracuse University), and Herb Hoffman; Greenstreet, Den Hartog, Alvin Weinberg (ORNL Director), and Hoffman; Roy Huddleston, Jim Corum, Sam Moore, Frances (Lamb) Moore, Ray Holland (University of Tennessee), and Joel Witt; Weinberg (banquet speaker) and Lyon.*

*In 1963 ORNL hosted the 2nd Southeastern Seminar for Thermal Sciences, with Herb Hoffman serving as conference chairman. Pictured from left are Hoffman, Jim Ferell (N.C. State University), Walter Frost (U.T. Space Institute), and George Lawson and F. Shahrokhi (U.T. Space Institute).*


*Beginning in the late 1960s the Reactor Division held an annual "Information Meeting on Studies in Applied Solid Mechanics." These meetings attracted relatively large numbers of national and international attendees to hear of ORNL and related work supporting design code development and structural design of various reactor concepts. Clockwise from top left are Delores Weaver accepting registration fee from Frank Williams, vice president of Taylor-Forge; Bill Greenstreet, Richard Gwaltney, and Pat Callahan; Mike Lundin; Weaver and Don Godwin at registration desk; John Brock, of the U.S. Naval Postgraduate School and Jim Robinson; Claud Pugh and Sam Moore.*

***The ORNL Heavy-Section Steel Technology Program, the Nuclear Regulatory Commission's large, long-running research program at ORNL, also held annual information meetings in the late 1960s and early 1970s. These scenes are from the fourth annual meeting in 1970.***

***In the outer pictures, clockwise from upper-left corner: attendees, including Neil Randall, TRW, third from left, and Larry Chockie, General Electric, far right, examine an HSST intermediate test vessel model. Sam Beall is presenting the feature address. Grady Whitman, left, and Bob Wiley, Southwest Research Institute, are third from left; Sue Freels, Grover Robinson, Bonnie Reesor, and Charlie Normand are at registration desk; two attendees examine fracture surfaces from 6-in.-thick HSST intermediate-size tensile specimens with part-through surface cracks; Door Doty, U.S. Steel, second from left, Steve Pawlicki, USAEC, third from left, and Gene Bailey, Commonwealth Edison, right, pick up copies of presentations; Beall, Harold Etherington, member of the Advisory Committee on Reactor Safeguards, and Floyd Culler, ORNL Deputy Director; Herb Corten, University of Illinois, and Ed Wessel, Westinghouse Research Laboratory, examine the test setup for a small flawed pressure vessel.***

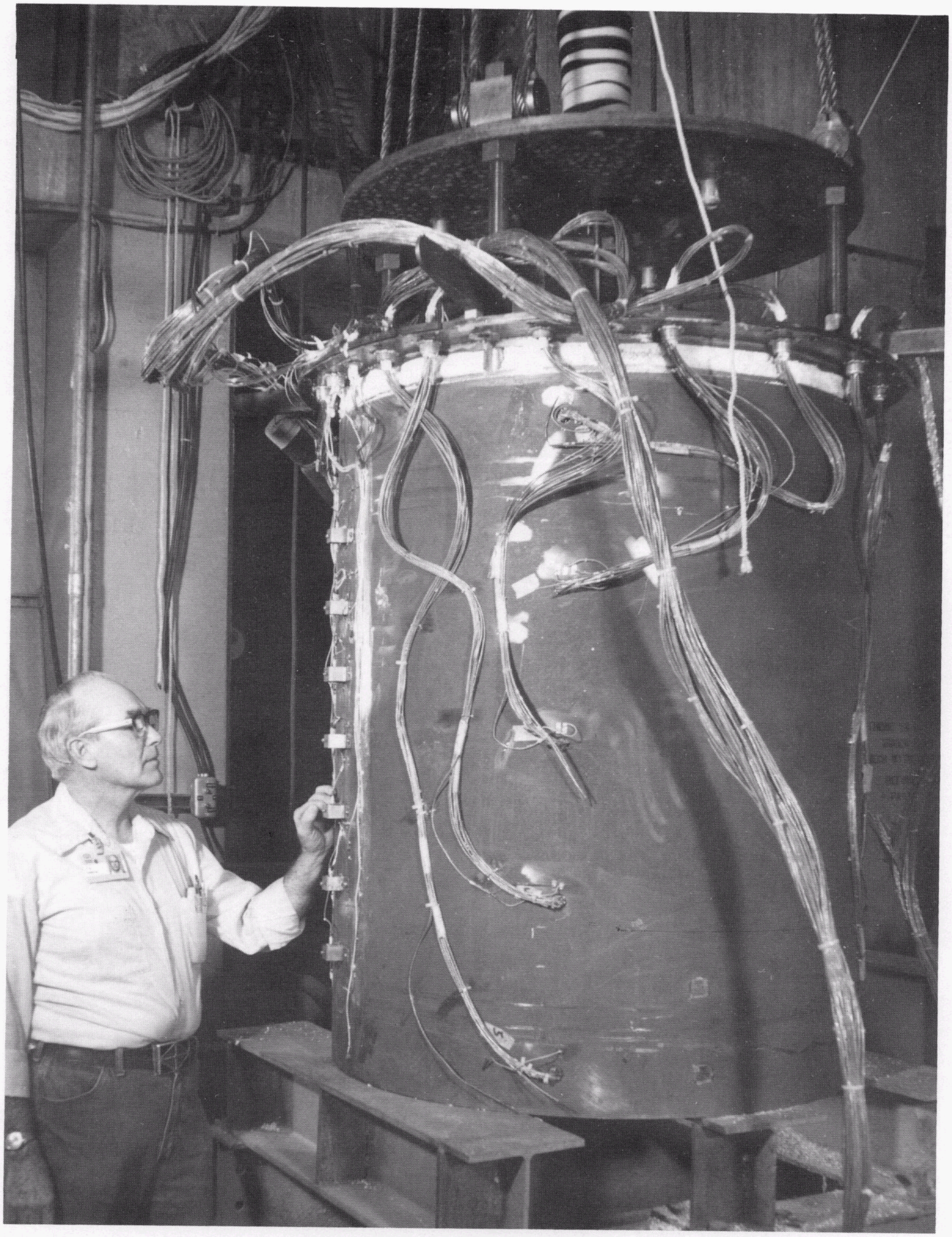
*Frequent interchanges with foreign investigators continued through the 1960s and 1970s. Here, from left, Chuck Preskitt, Bud Perry, and Dick Cheverton talk with Nenad Raisic (third from left), Head of the Reactor Physics Department, Institute of Nuclear Science, Belgrade, Yugoslavia, in 1963.*


*In the 1960s it became apparent that the Reactor Division was accumulating an enviable safety record—no lost time accidents since the first ORNL reactor personnel moved to Y-12 in September 1950. In 1964, Rodger Hibbs, Y-12 Plant superintendent, who later became president of Union Carbide Nuclear Division, presented a plaque to the Reactor Division recognizing 14 years of safety. From left are H. G. MacPherson, ORNL Deputy Director and former acting division director; Hibbs; Bob Helton, division safety coordinator; Sam Beall, Division Director; and Will Osborn, head of division Engineering and Administrative Services.*

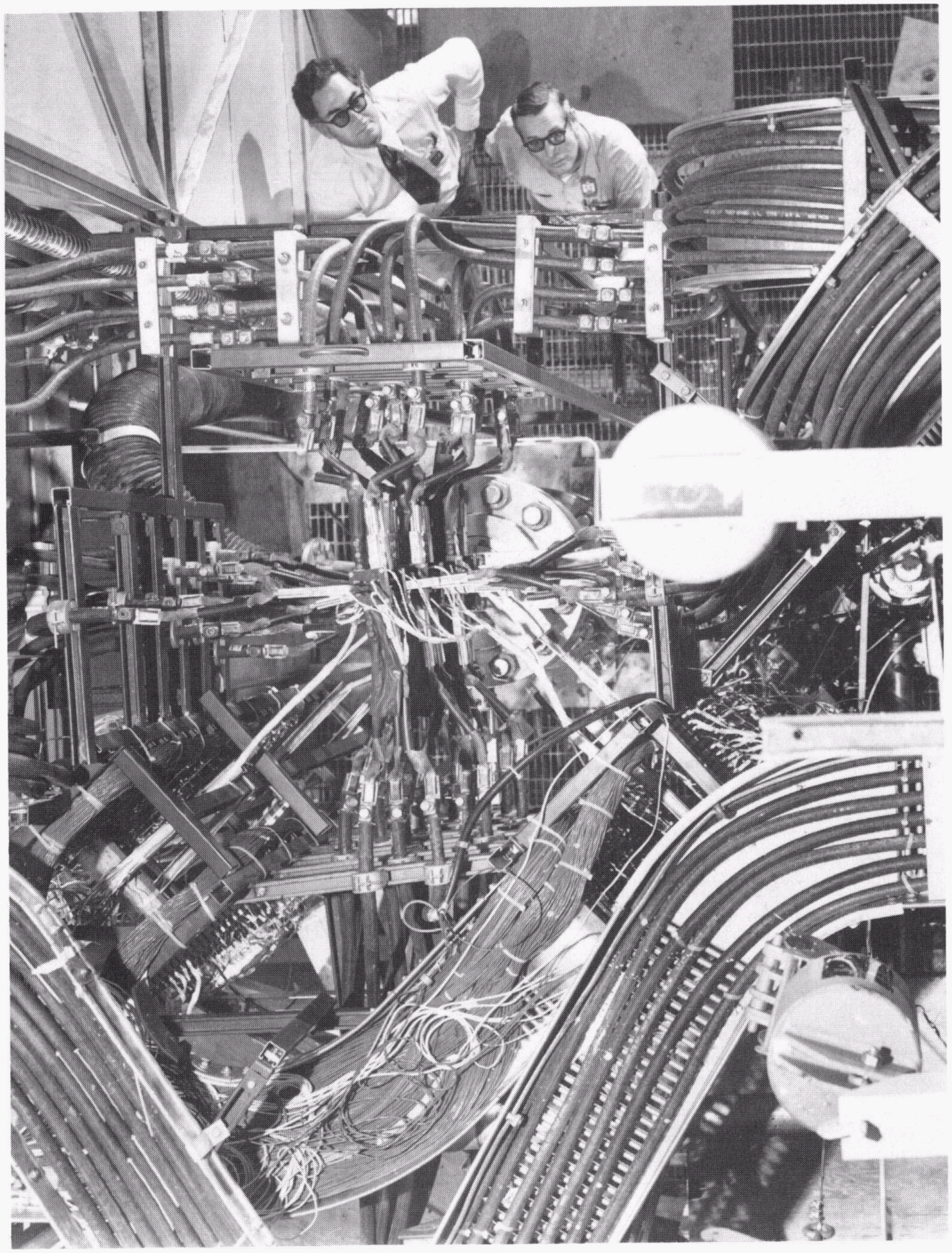
***The Nuclear Safety Information Center staff gathered for this 1967 picture. From left are Gerry Keilholtz, Eugene Cramer, Harry O'Brien, Clint Walker, Reeta Fletcher, John Merkle, Paul Blakely (partially hidden), Jeannie (Thomas) Scott, Mario Fontana, Celia Murphy, Becky Wallace, Bill McClain (partly hidden at back), Howard Whetsel (partly hidden), Don Jacobs, Mel Winton, Dianne Lane, Tom Lomenick, Bill Ergen, Ray Scott, and Joel Buchanan.***

*Celebrations for reaching 25 years of company service became increasingly common in the late 1960s and early 1970s, as typified here. Left column from top are Mike Lundin, Bernice Fitzgerald, Bob Cauble, and Margaret Wilson; center column are Charlie Mills, Ray Clark, Bob Smith, and Charles (Chigger) Wallace; and right column are Malcolm Richardson, Woodrow Terry, and John Tudor.*

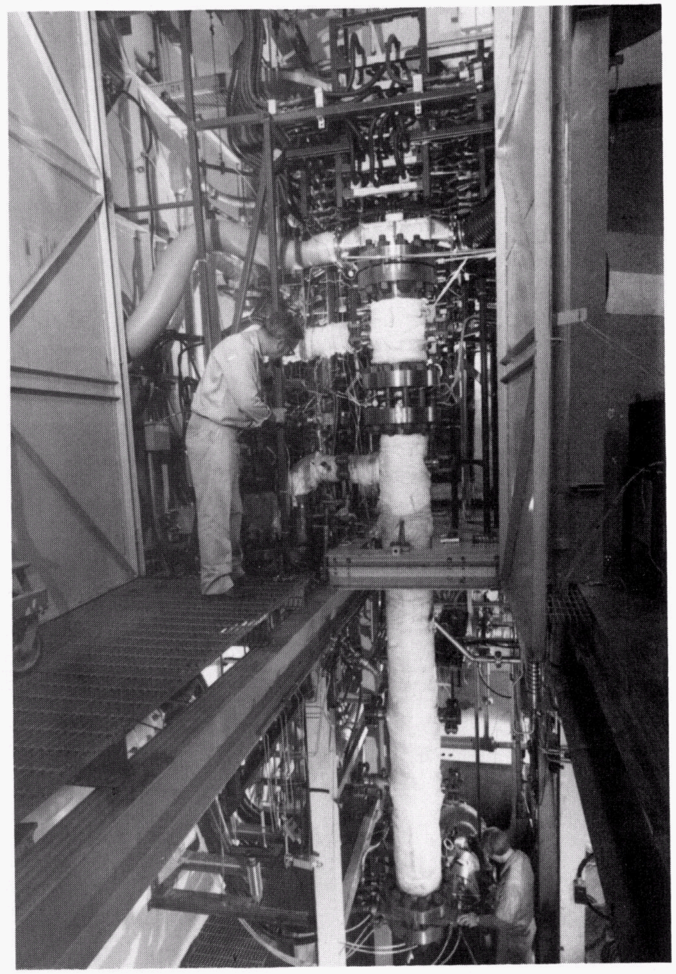
*Art Fraas reached his 25th anniversary in 1975. Clockwise from top left are (left to right) Mac Lacky, Murray Rosenthal, Fraas, Don Trauger, Sam Beall; Jean White, Art Miller, John Moyers, Fred Lynch, Bob Holcomb, Bill Mixon, Jeannie Scott, John Clarke, and Gordon Fee; Bill Montgomery, Virgil Haynes, and Paul Gnadt; Dick Lyon, Marty Lubel, Bill Greenstreet, Lynch (partially hidden), Fraas, and Scott; Trauger, H. G. MacPherson, Fraas, Bud Perry, and Beall.*



***T. A. King inspects instrumentation on a cylindrical shell to be subjected to a thermal-shock test in the Heavy-Section Steel Technology Program. The thermal shock is produced by using liquid nitrogen to rapidly cool the inside surface. This produces conditions similar to those that exist when emergency coolant water comes in contact with the hot inside surface of a reactor vessel following a hypothetical loss-of-coolant accident (LOCA).***

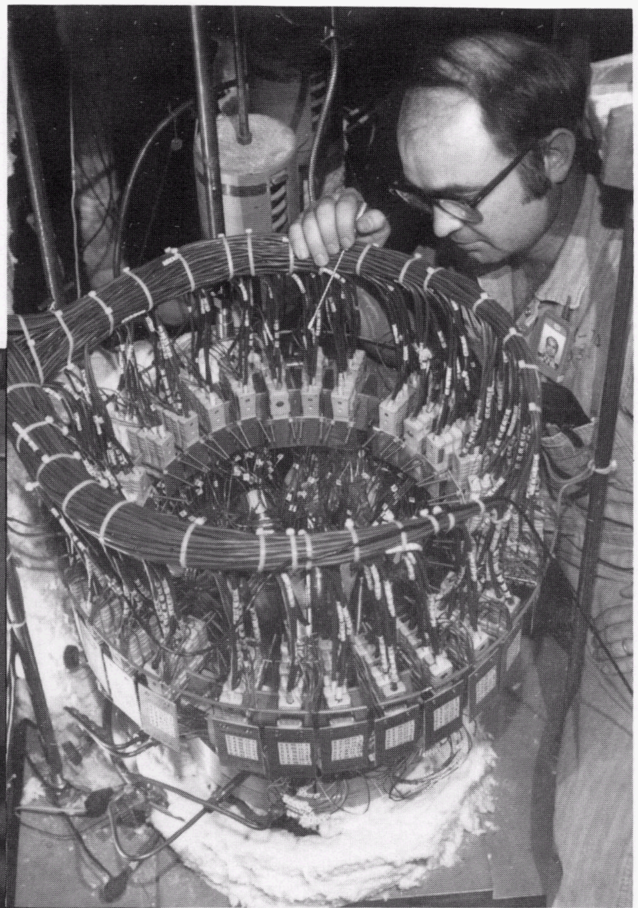


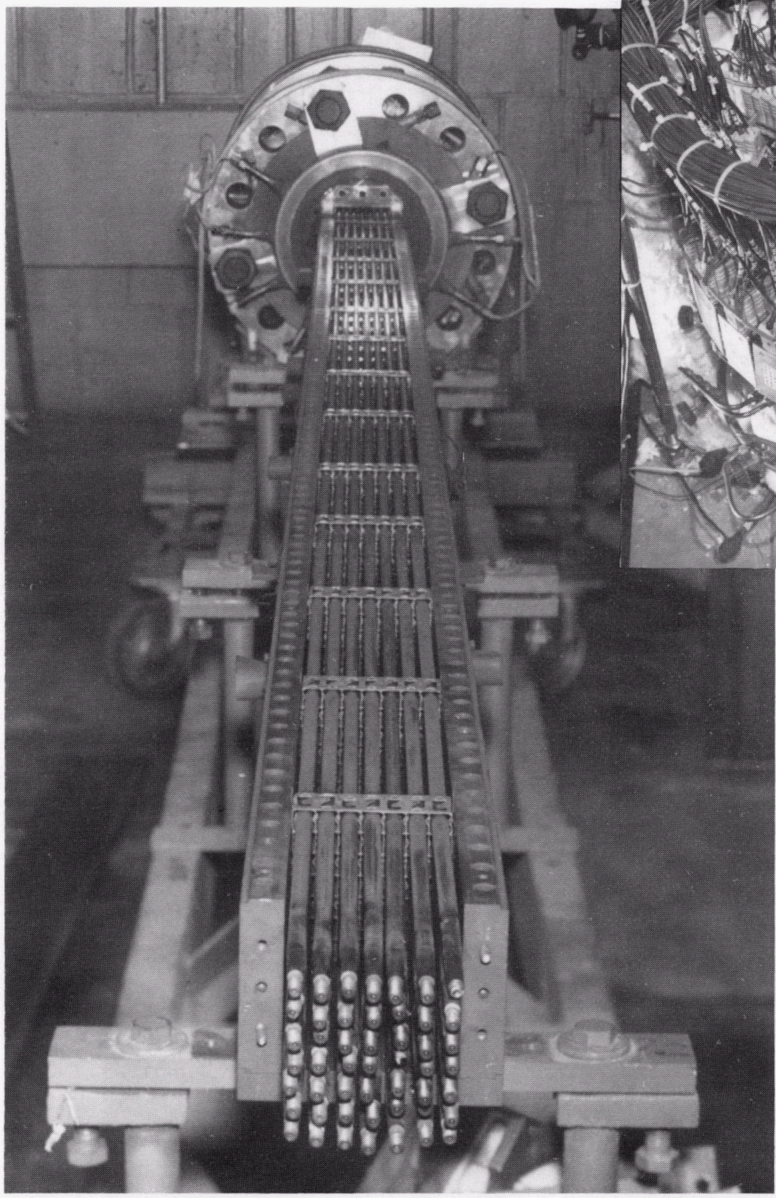
*Overhead view of core vessel of the Thermal Hydraulic Test Facility (THTF) illustrating complexities involved in attaching electric power connections to the fuel rod bundle. Clockwise on the right are J. L. Crowley and J. E. Wolfe. The THTF was used to study temperature, pressure, and coolant flow conditions during a severe LOCA in a pressurized-water reactor, an event known as "blowdown."*



***Elevation view of THTF.***

*Closeup view of power and instrument leads for the fuel assembly simulator.*



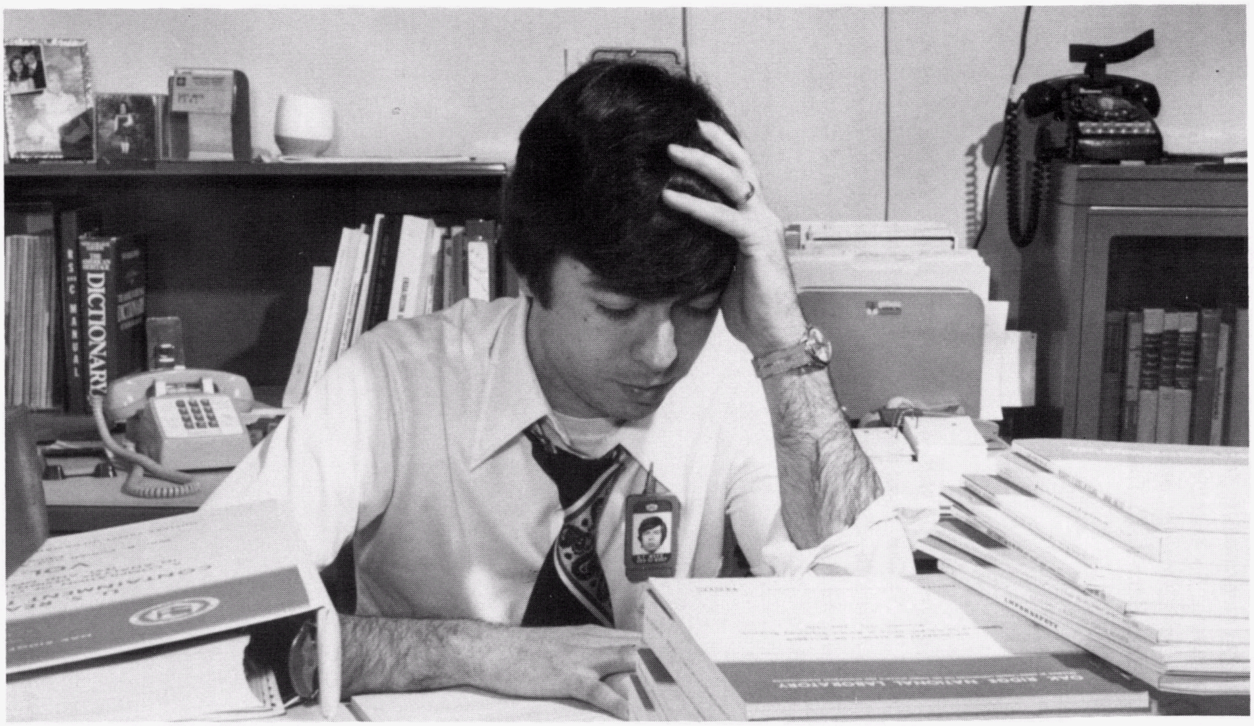


*Fuel rod bundle simulator used in THTF. These fuel assembly simulators were designed and fabricated in the division.*

***J. L. Crowley examining a fuel pin simulator bundle. The simulators, in this case, consisted of an electric heater element inserted into a 5-ft-long Zircaloy tube pressurized with helium gas. These bundles were used to investigate swelling and bursting of fuel rods during a reactor LOCA.***

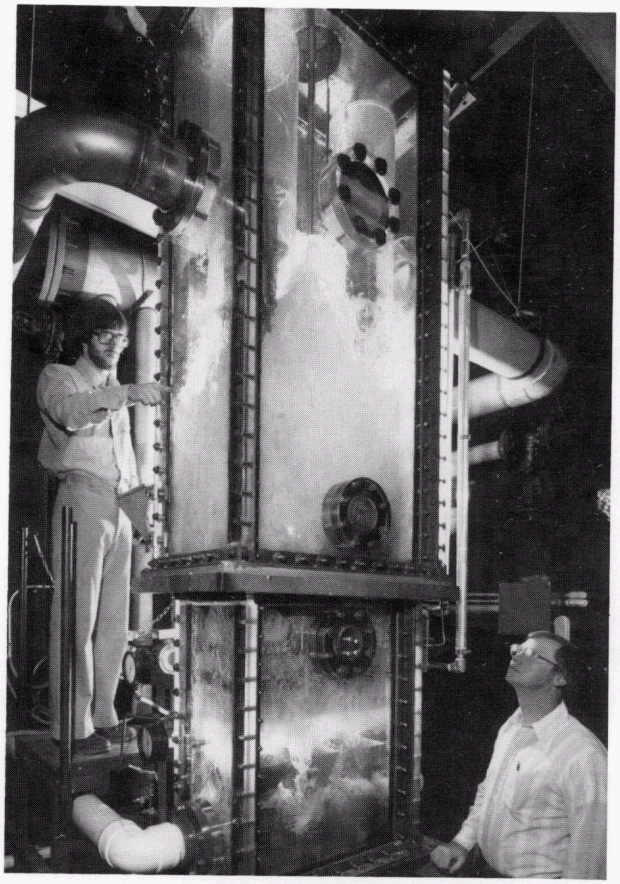

*Nuclear Safety celebrated its 30th anniversary in 1989. It continues to be a valuable resource for reactor designers, builders, and operators and for researchers, administrators, and safety officials in both government and private industry. Here J. E. Jones congratulates J. R. Buchanan on the achievement.*

*The Nuclear Safety staff is shown in this 1978 photo. In the front row, from left, are Lisa Nation, Joan Roberts, William Cottrell, Angie (Puckett) Redford, Walter Jordan, Myrtleen Sheldon, and Ann Ragan. In the back row are Joel Buchanan, Paul Haas, Ed Hagen, Herschel Godbee, Ed Compere, Rowena Chester, and Owen Hoffman.*

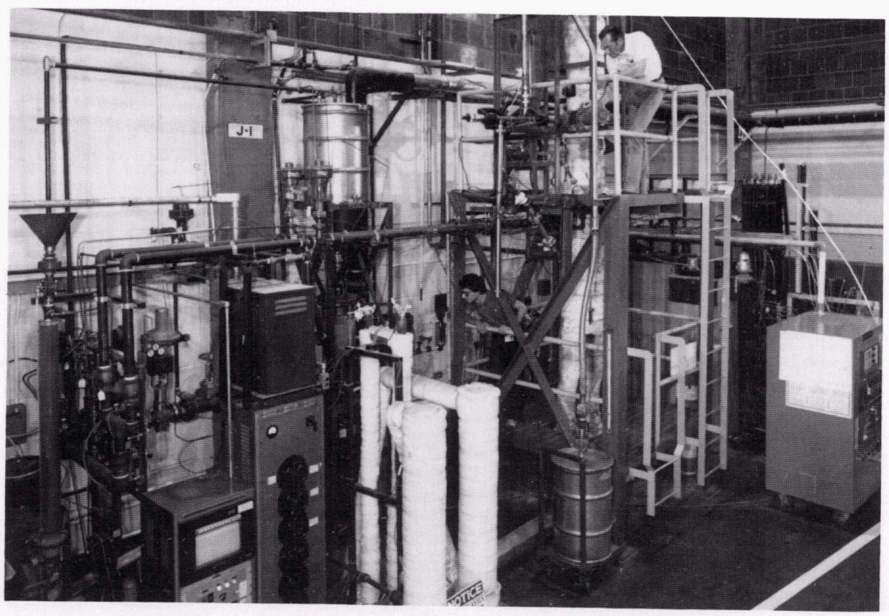



*G. T. Mays is shown in this 1979 picture in the Nuclear Safety Information Center where staff members abstracted pertinent articles and reports published in technical journals, issued by research laboratories or presented at technical meetings.*

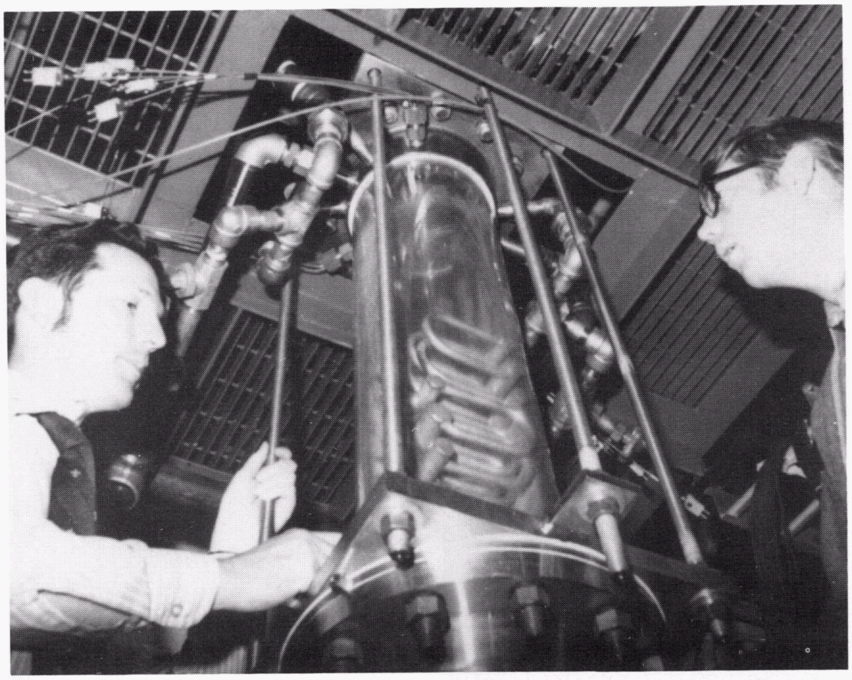
*Staff members at the Nuclear Safety Information Center placed some 12,000 documents into its data storage bank each year. Shown are P. G. Cleveland, left, and D. S. Queener.*



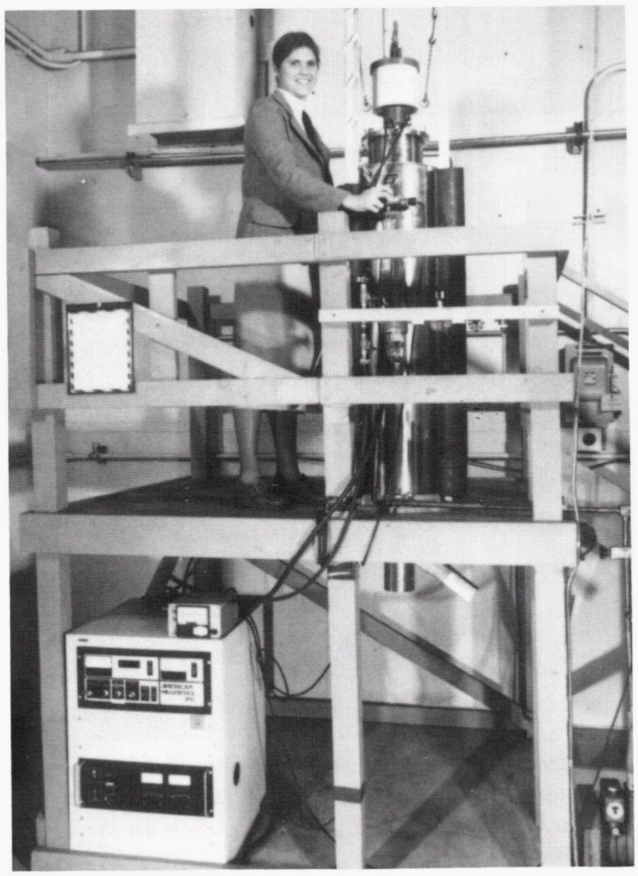
*Instrument development loop used to simulate simultaneous movement of steam and water through a reactor during the water injection portion of a LOCA. This air and water test assembly provided data for developing, testing, and calibrating measuring instruments. Shown are J. E. Hardy, left, and S. K. Combs, right.*



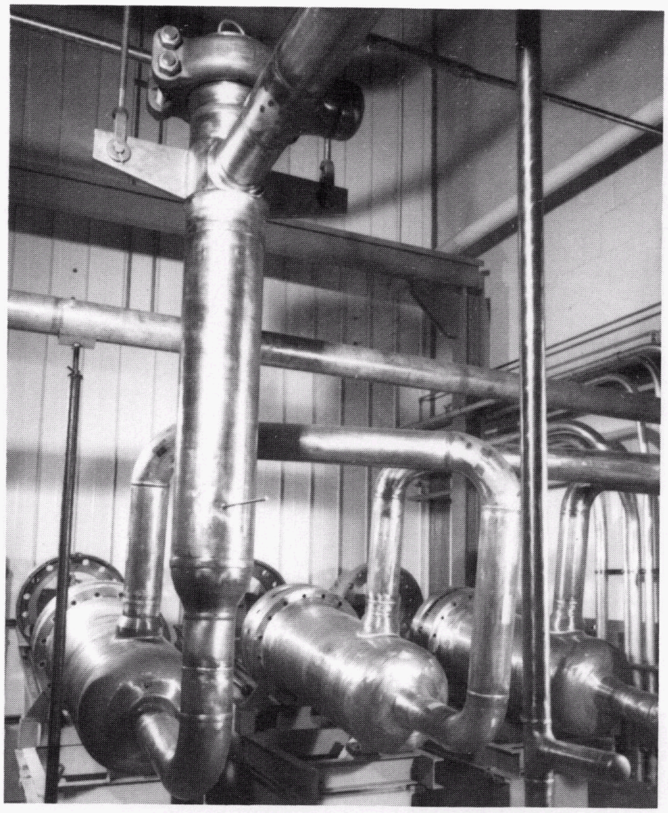

*Fluidized bed coal combustor instrumented for flue gas diagnosis. Coal was burned in a fluidized bed containing limestone to demonstrate the sulfur dioxide capturing capability of the latter in this application. G. P. Zimmerman and F. E. Lynch are shown on the lower and upper platforms, respectively, of the structure surrounding the combustor.*



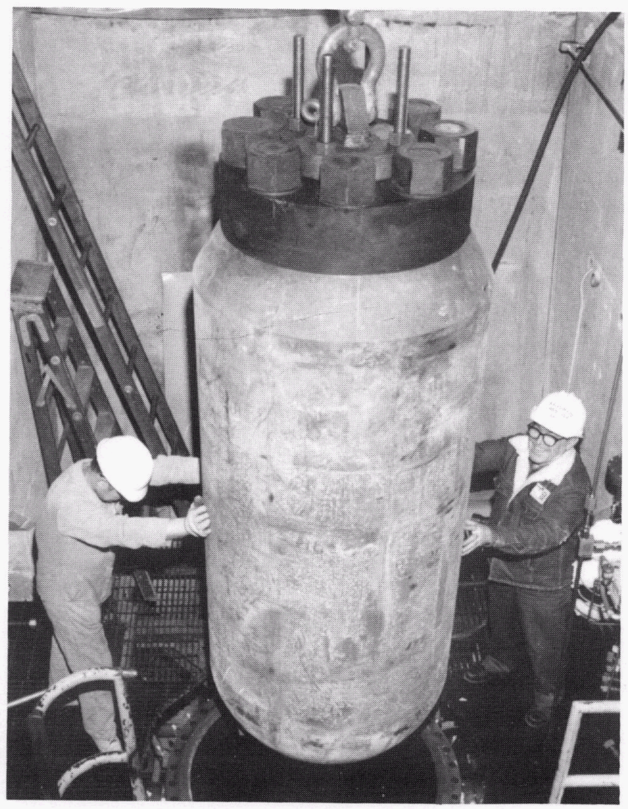
*R. S. Holcomb, left, and J. E. Jones, right, examining the laboratory-scale fluidized bed combustor for burning coal.*



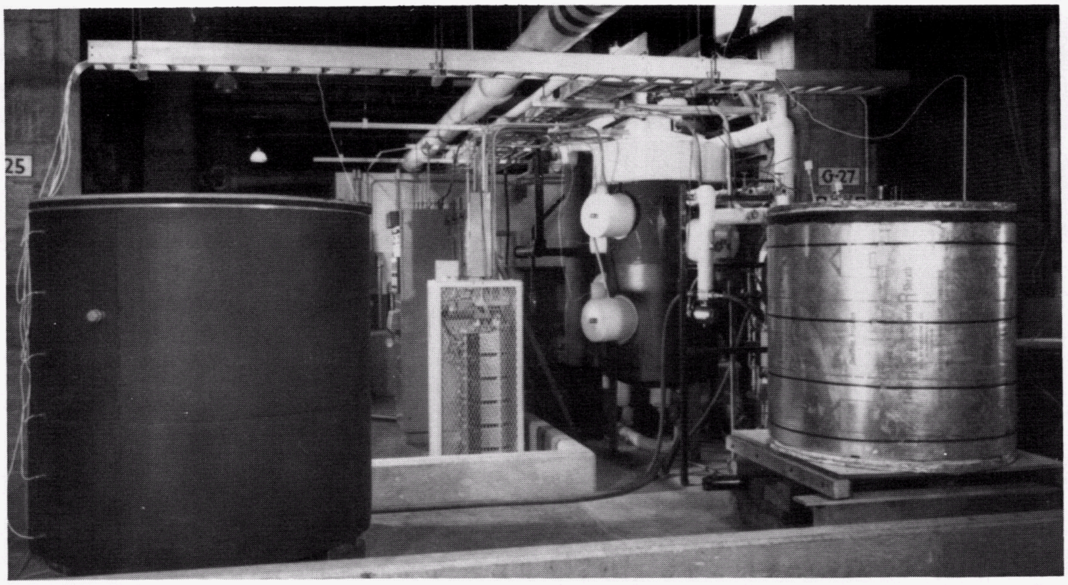
*D. L. Mailen standing alongside the open gradient magnetic separation facility for beneficiation of coal and other minerals. In this facility, a superconducting magnet separates mixtures of solids into multiple streams.*



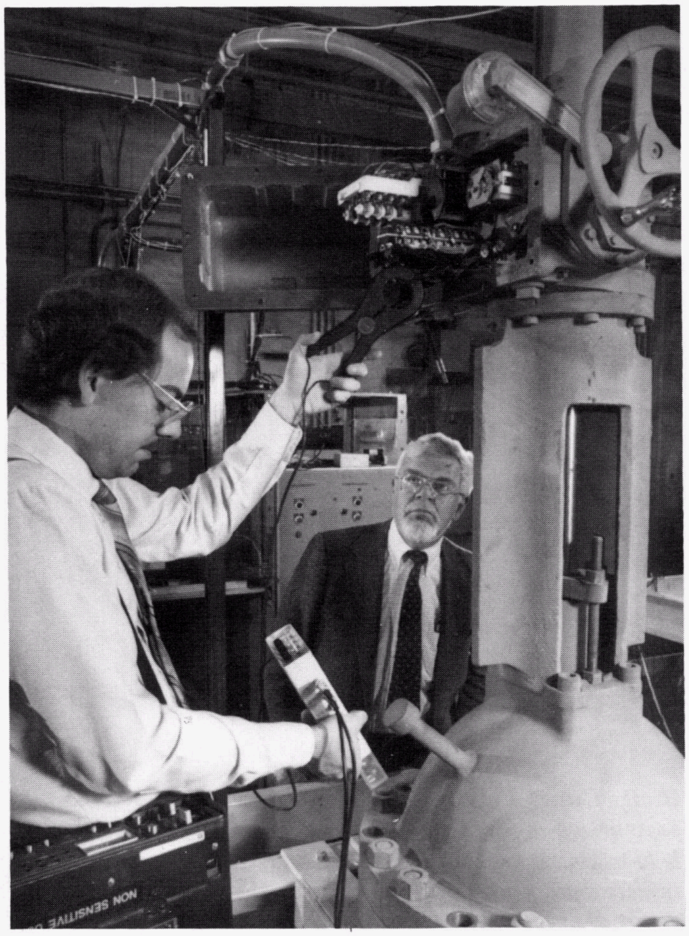
*Gas-bearing circulators in the Component Flow Test Loop (CFTL) provide pressure to simulate the flow conditions present in helium-cooled power systems. The 20-kW, variable-speed motors circulate helium at temperatures up to 1835°F and pressures of 5000 psi. The CFTL makes it possible to evaluate the in-service performance of power system components in high-temperature environments.*



*H. D. Curtis helps to install a test vessel for a pressurized-thermal-shock experiment. The purpose is to examine the influence of the safety system that injects cold water into the pressurized pressure vessel of an operating reactor during a LOCA.*

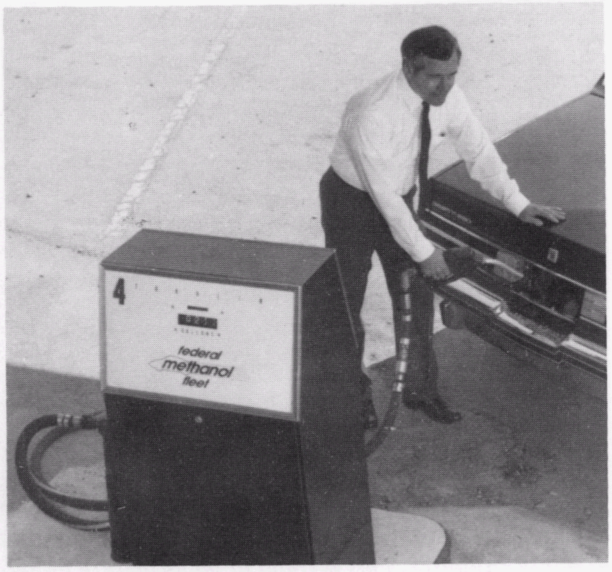



***This versatile test facility provides the capability for performance testing thermal energy storage devices at temperatures up to 125°F for small building applications. Among the storage system variables studied in this facility are energy storage capacity, efficiency, and input and output heating rates.***

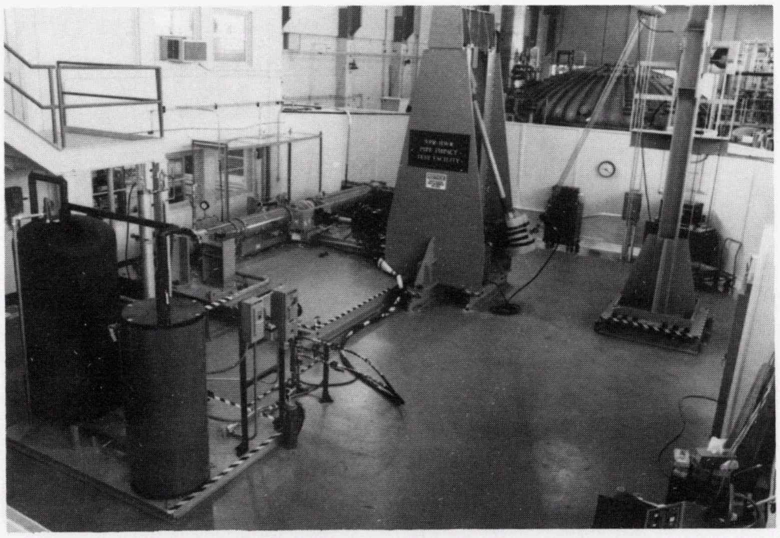



***H. D. Haynes and D. M. Eissenberg obtaining and recording electric current signals during motor-operated valve operation. The resulting motor current or power signal can be used to detect and assess degradation.***

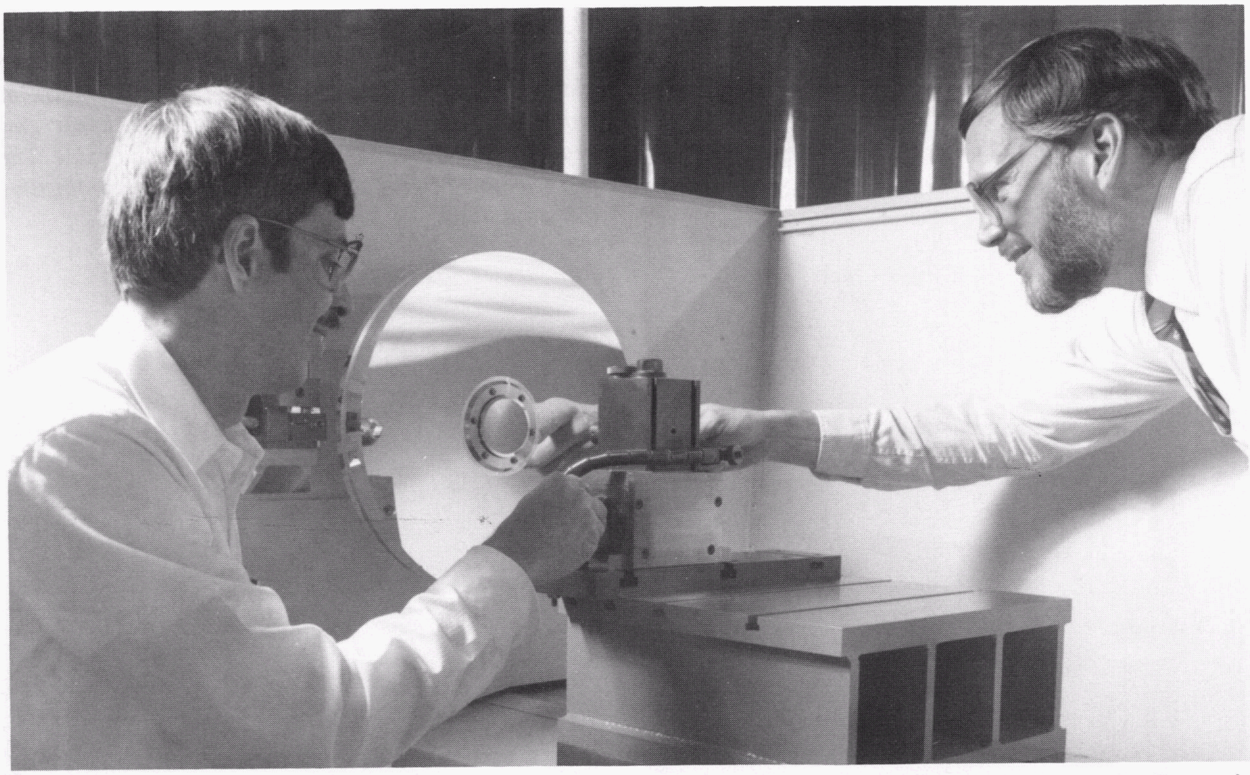
*W. K. Kahl inspecting a Zirconia-coated piston crown in a methanol-fueled research engine (on left side of picture with head removed) after several hours of running. The Zirconia-coated insert with air gap provided a thermal barrier at the combustion side. The test was to examine surface influence on combustion and emissions.*



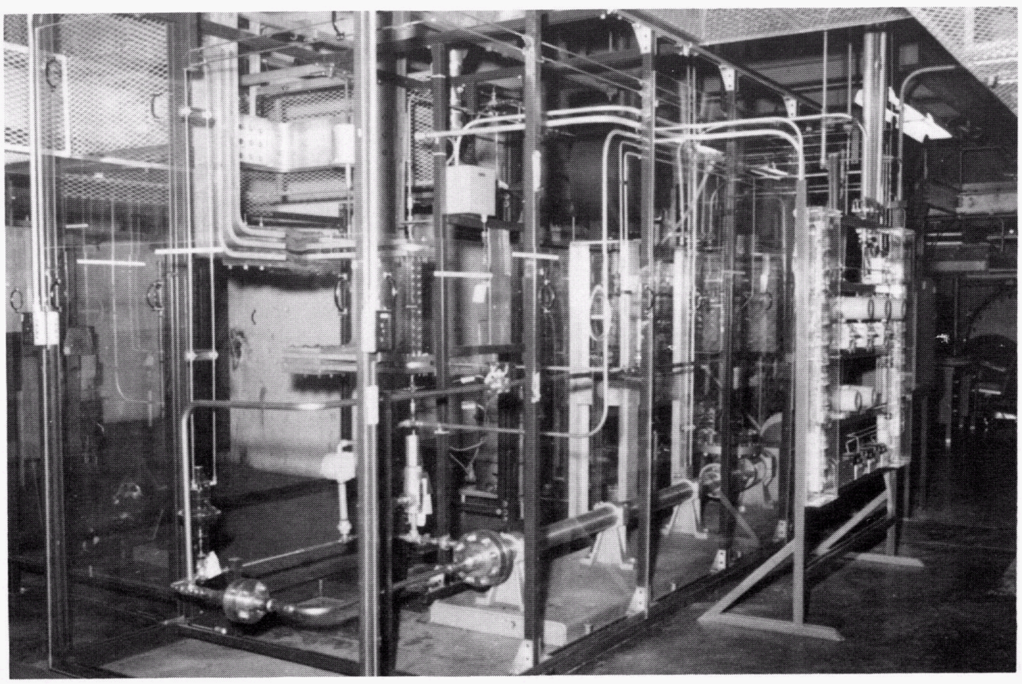

*R. N. McGill is shown refueling one of the vehicles from the methanol-fueled fleet used to obtain performance data as a part of the Alternate Fuels Utilization Program.*



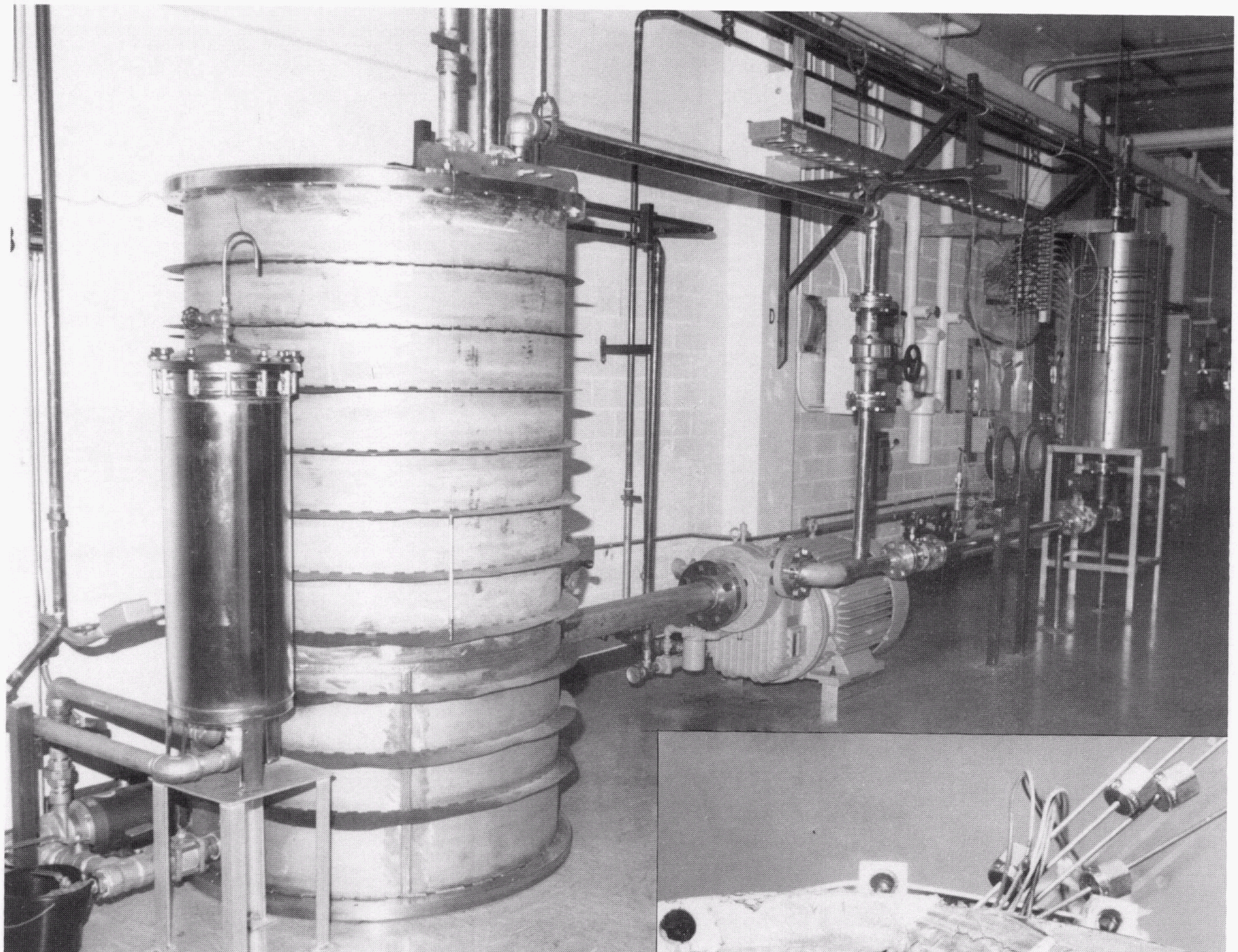
*The role of this pipe impact facility is to test the validity of requiring reactor primary coolant pipe to be designed for an "instantaneous" double-ended-guillotine-break. Use of the facility has demonstrated that such an event is highly improbable for austenitic stainless steel pipe to be used in the New Production Reactor/Heavy Water Reactor.*

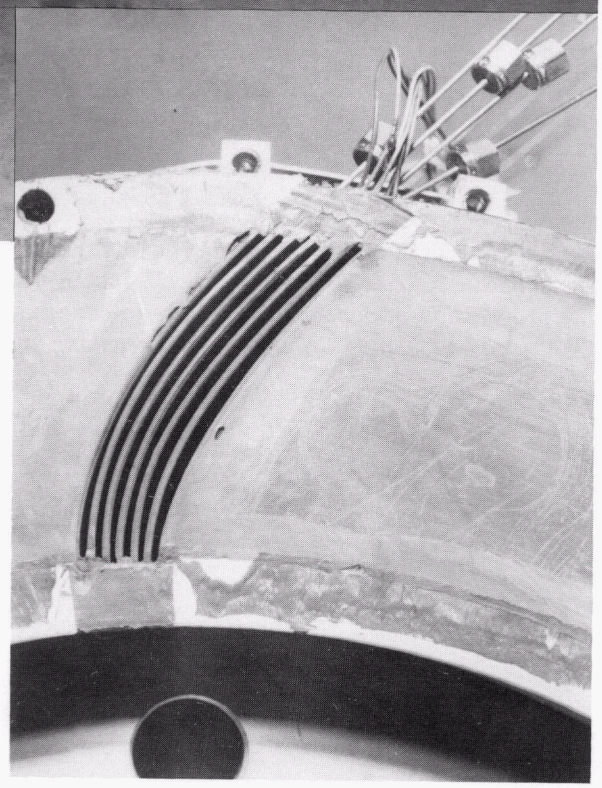


***J. P. Cunningham and A. C. Miller monitoring diamond, single-point turning of a mirror for the Strategic Defense Initiative. This method of manufacture produces an extremely high-quality product.***

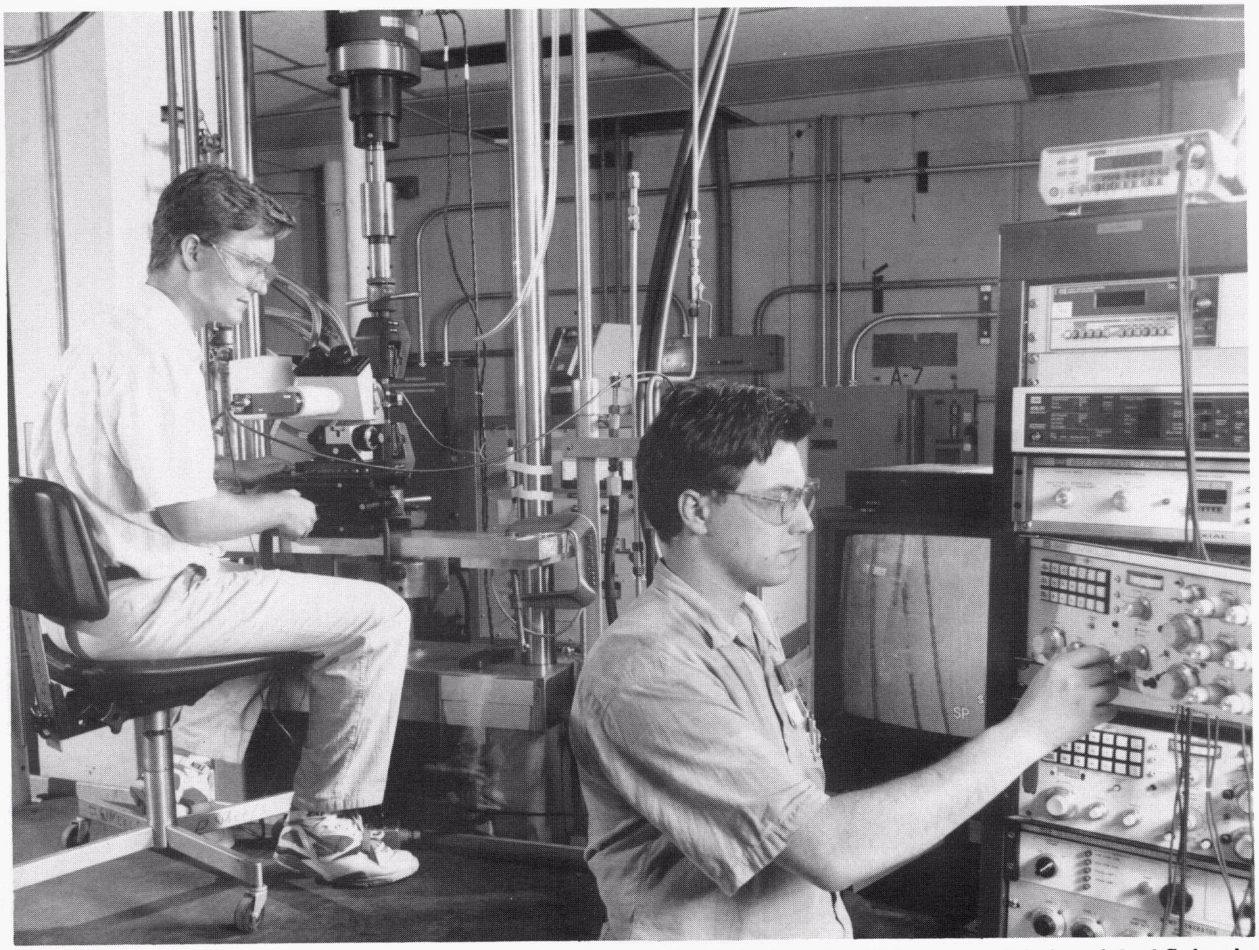


***Building on capabilities developed in connection with light-water reactors, a Thermal-Hydraulic Test Loop is operated in support of the Advanced Neutron Source (ANS) design and evaluation activities. The ANS is to be an advanced irradiation facility that will supersede the HFIR.***



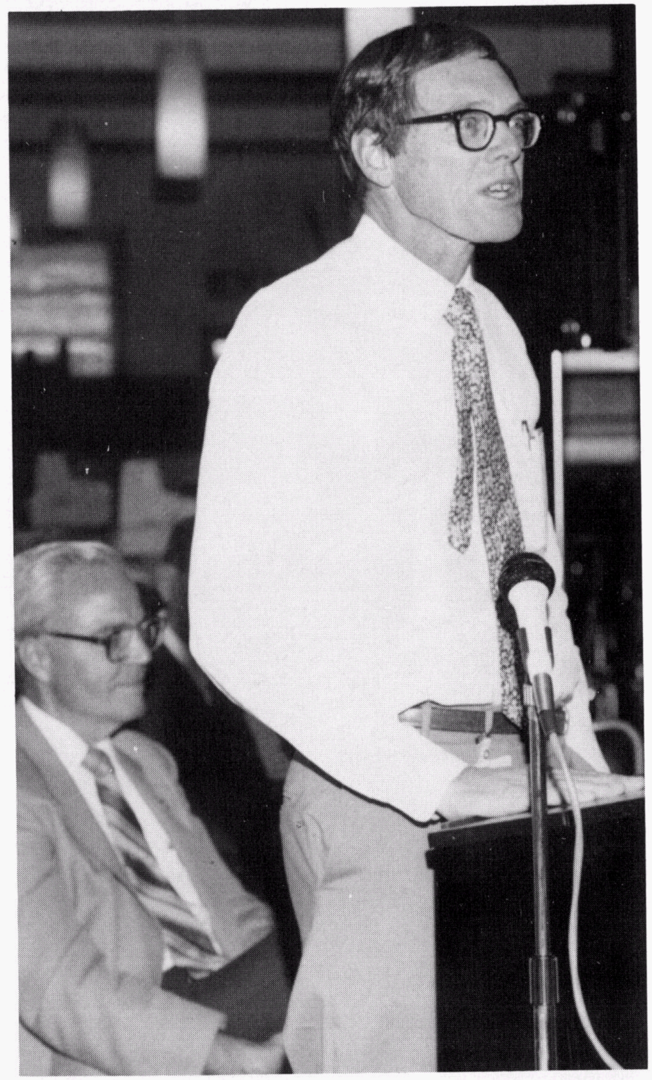


*ANS fuel plate mechanical stability under design flow conditions is examined in the Fuel Plate Stability Test Facility shown. Shown in the inset is an epoxy test assembly with a section of fuel plates that are curved in the form of an involute. The objective is to ensure that uniform spacing is maintained between plates.*



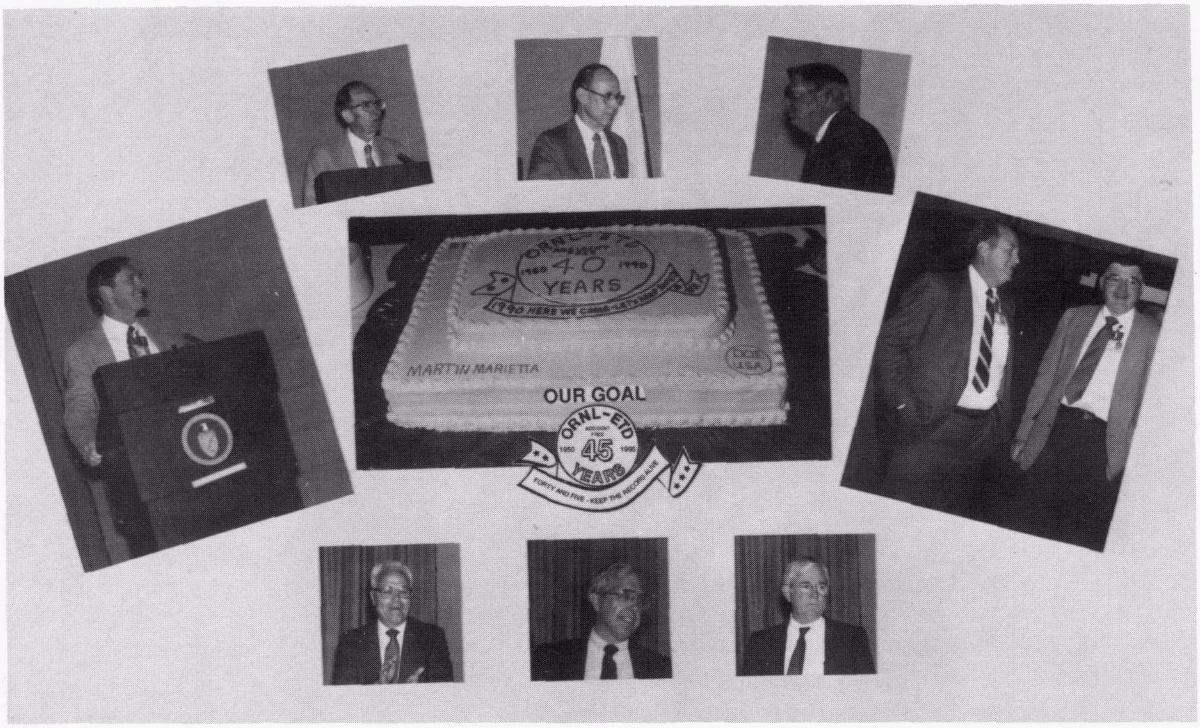

***Engineering Technology Division has a jointly appointed ORNL/University of Tennessee Distinguished Scientist in composite materials and structures, Y. Jack Weitsman. Weitsman's research in ETD is focused on modeling of the progression of damage and failure in continuous fiber ceramic composites. Here, graduate students Brett Okhuysen and Donald Erdman observe cracking in a composite specimen during a tensile test.***


***By the end of 1979, the Engineering Technology Division's safety performance record had reached an almost unprecedented 29 years without a lost-time injury. Jack Case, Y-12 Plant superintendent, presents Herb Trammell (left), Division Director, and Don Trauger, ORNL Associate Director, with a plaque as Clarence Johnson (right), Y-12 safety head, looks on.***



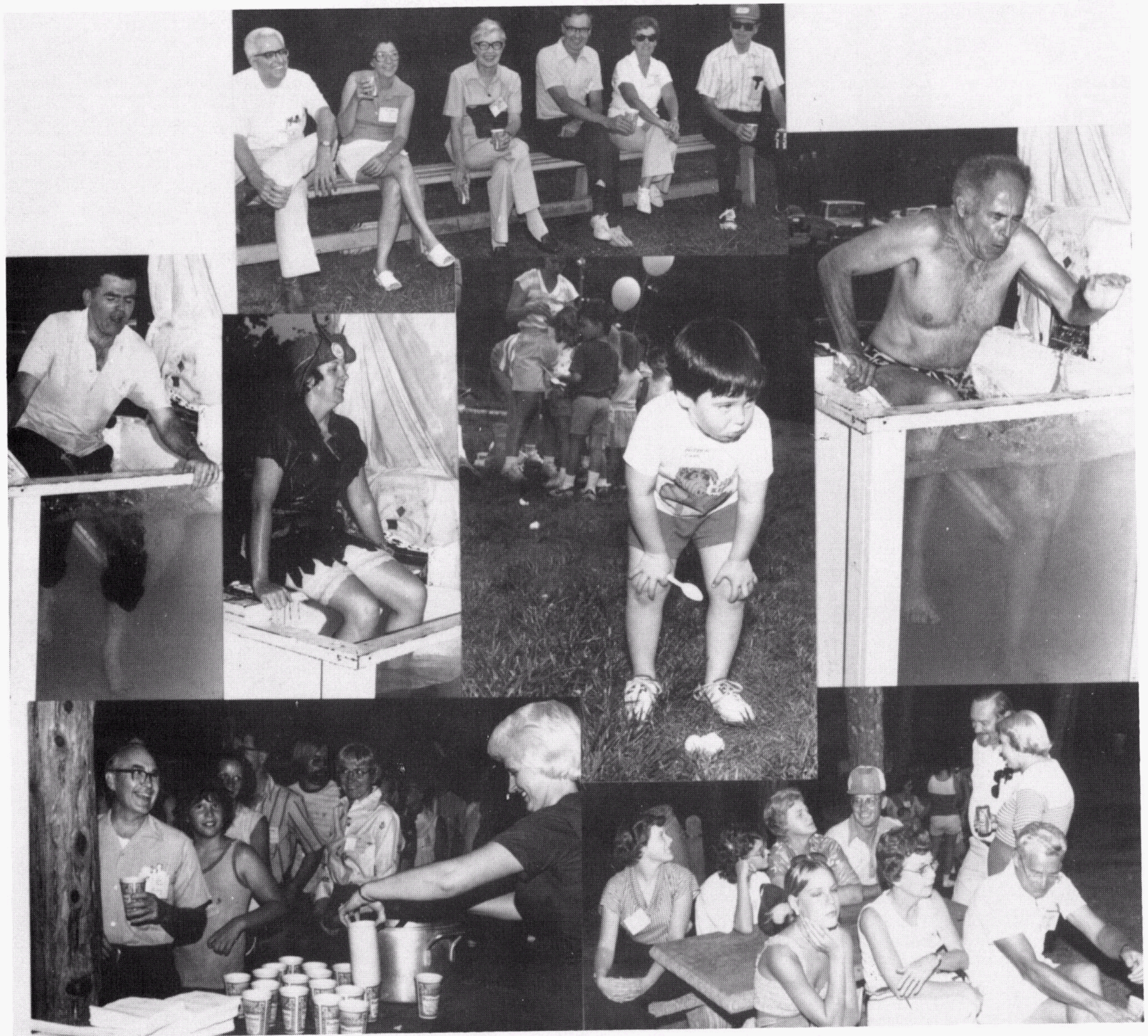
***In 1985 the division reached another safety plateau—35 years without a lost-time injury. At left, Herb Trammell talks about the division's 35-year record as Gordon Fee (facing camera at left), Y-12 Plant superintendent and former division director, and Clyde Hopkins (with side to camera), Martin Marietta Energy Systems Senior Vice President, looks on. At right, Herman Postma, ORNL Director, congratulates the division.***


*Safety meetings and innovative safety contests helped to educate division staff and heighten safety awareness, thus contributing to the continuing safety record. Charlie Mills, division safety coordinator, stands with safety slogan contest winners in this 1982 picture. The winners, from left, are Bob MacPherson, Amy Leslie, Louise Bible, Judy Kibbe, and George Lawson.*




*September 1990 arrived with the division's safety record intact—40 years without a lost-time injury. To mark the event, a party was held for the entire division. Martin Marietta Energy Systems and Department of Energy dignitaries as well as former division directors turned out to join in the celebration. Those making remarks included, clockwise from left, Herman Postma, Energy Systems Senior Vice President; H. G. MacPherson, former division director; Don Trauger, senior staff assistant to the ORNL director; John Jones, Division Director; Clyde Hopkins, Energy Systems President and Gordon Fee, Y-12 Plant Superintendent and former division director; Alvin Trivelpiece, ORNL Director; Jim Reafsnyder, Deputy Assistant Manager, Energy Research and Development, DOE Oak Ridge Field Office; and Herb Trammell, former division director.*

*Part of the crowd at the division's 40-year safety celebration enjoys the reception that followed.*



***Division-wide picnics were regularly held in the 1970s, and the tradition was picked up again in 1987 and 1992. Except for the unhappy lad whose egg broke during an egg tossing contest (1987), these scenes are from the 1977 picnic. The dunking booth was a popular attraction in those days. Pictured taking their turn in the water are, from left, Gordon Fee, Sharon Fuller, and Irv Spiewak.***


***In 1986 the Engineering Technology Division held a "Fall Fling," which was a cruise and dinner aboard Knoxville's River Queen. The highlight of the evening, at least for Don Burton, is shown in the center picture. He was chosen to be the "center of attention" for the guest bellydancer.***

*Twenty-five year service awards continued to be benchmark events under Union Carbide. When Martin Marietta became the operating contractor, service awards were provided every 5 years. Here, Dan Curtis is shown with members of the Solid Mechanics Section after receiving his 25-year award in 1977. Pictured from the right in the front row are Bill Greenstreet, Grady Whitman, Curtis, Jim Corum, Delores Weaver, Susan (Carr) Jennings, Tom King, and Don Godwin. Second row: Jack Smith, Grover Robinson, Sam Moore, Terry Yahr, Sam Bolt, and Hubert Guinn. Third row: Joe Blass, Richard Gwaltney, Bob Bryan, Charlie Hurtt, and Howard Butler. Fourth row: John Clinard, Dave Robinson, Wallace McAfee, John Bryson (partially hidden), Malcolm Richardson, and Gordon Smith. Fifth row: Tom Hill.*

*In 1979, Ernie Silver received his 25th anniversary award. Helping celebrate, from left, are Lincoln Jung, Frank Zapp, Terry Yahr, Richard Gwaltney, Fred Hannon, Alex Zucker, Gerard deSaussure and Wallace McAfee (partially hidden), Rafael Peiez, Bob Peele, Mary Phillips, Jim Corum, Sam Bolt, Ruth Nesbitt, Fred Maienschein, John Clinard (partially hidden), Don Steiner, Susan (Carr) Jennings (partially hidden), Joe Blass, Florence Olden, Delores Weaver, and Sue Freels.*


*John Wolfe (in suit) reached his 25th anniversary in 1981. Pictured from left are Ken Finnell, Ralph Dial Wolfe, Jack Money, Hardin Duckworth, Howard Freeman, Cleois Cross, Ed Biddle, and Tom Wynn.*

*Marselle Ruszkowski celebrated her 25th anniversary in 1982. Pictured with her are (front row from left) Jill Smith, Margaret Wilson, Steve Hodge, Grady Whitman, and Herb Hoffman; (back row) Sue Freels, Pete Carlson, Pat Chaffee, Lou Parsly, John Moyers, Van Brantley, and Tom Dahl.*

*Harry Young received his 25th anniversary certificate in 1983. From left are Al Grindell, Uri Gat, John Sanders, Ernie Lees, Dick Huntley, Virginia Maggart, Young, and Ed Biddle. Ron Senn is partially hidden at right.*

***Herb Trammell received his 40-year service award from Alex Zucker, ORNL Associate Director, at the last division staff meeting before his retirement in 1989. Looking on from left are Dan Naus, Bill Snyder, Fay Duncan, Don Burton, Joel Buchanan, Larry Jordan, Sharon Mashburn, Terry Yahr, Steve McNeany, and Ted Fox.***

***In recent years, individuals receiving long-time service awards often elected to receive their award from the Laboratory Director. Clockwise, from above left: Terry Yahr receives his 30th from Alvin Trivelpiece as John Jones, Alex Zucker, and Jim Corum look on; Ernie Silver receives his 35th as Jones and Zucker look on; Mel Tobias receives his 40th as Ted Fox, Jones, and Zucker look on; and Tom Hill receives his 35th from Herman Postma as Rick Battiste and Corum look on.***

*Retirements occurred with increasing regularity during the 1970 to 1992 period. Solid Mechanics Section personnel gathered in 1976 to wish Hugh MacColl a happy retirement. From left: Dave Robinson, Richard Gwaltney, John Clinard, Claud Pugh, Delores Weaver, Linda Dockery, John Bryson, Jim Corum, Don Godwin, MacColl, Susan (Carr) Jennings, Sam Moore, Joe Blass, Terry Yahr, Jack Smith, Terry Delph, Pete Holz, Sam Bolt, Wallace McAfee, Hubert Guinn, and Jay Clairborne.*

*Dick Lyon was joined by his wife, Bobbie, at his 1976 retirement.*


***Dave Clark and happy group look on as Stan Ashton is presented a money wreath by Jim Teague at his retirement in 1978.***

*Delores Eden has a money tree in this 1982 retirement scene. She is joined by former bosses, (from left) Dick Lyon, Herb Hoffman, and John Michel. Energy Division director, Bill Fulkerson, is at right.*

*The Engineering Technology Division held a joint retirement party in 1985 for Marselle Ruszkowski and Bob MacPherson. Marselle is pictured with former boss, Will Osborn; while Bob is shown with secretary, Bettye Seivers.*

*This group gathered on the occasion of John Conlin's 1985 retirement. From left are John Wantland, Bob Adams, Al Longest, Martin Grossbeck, Mel Tobias, Norbert Chen, unidentified, Simon Rose, Ken Thoms, John Wolfe, Ira Dudley, Dave Lloyd, John Petrykowski, Bettye Seivers, Dennis Heatherly, Chigger Wallace, Harry Young, Earl Clemmer, Bill Nelson, Conlin's daughter, Kathy Rosenbalm, Bill Montgomery, Conlin, Bob MacPherson, Roberta Poe, Colin West, Jim Crowley, Virginia Maggart, and Ilana Siman-Tov.*

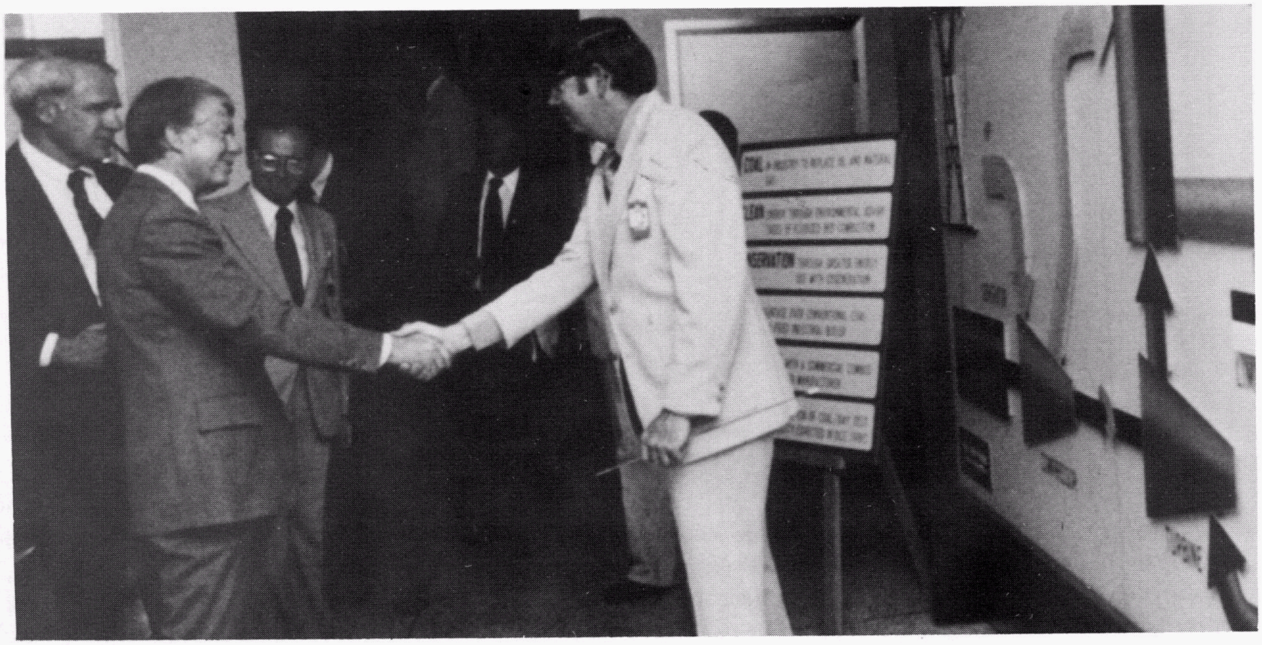
*Frank Zapp began his Oak Ridge career in the war years. When he retired in 1988, he was joined by a large group of current and former coworkers. First row, from right: Ruth McKee, Zapp, Ray Hudson, Yukio Takahashi, assignee from CRIEPI in Japan. Second row: Charlie Mills, Jim Corum, John Clinard, Marty Marchbanks, Lincoln Jung, Joe Blass, Angie Freeman, Barbra Booker. Third row: Dan O'Connor, Richard Gwaltney, Claire Luttrell, Julie Robinson, Delores Weaver, Linda Dockery, Florence Olden, Jean Fraley. Fourth row: Herb Trammell, Art Fraas, Harry Moseley, Barry Oland, Dan Naus, Grover Robinson, Perk Cooper. Fifth row: Chuck Claffey, Charlie Collins, Shih-Jung Chang, Roy Huddleston (partially hidden), Sam Bolt, Jay Clairborne, Moshe Siman-Tov. Back row: Sam Moore, Terry Yahr, Yung-Lo Lin, Rick Battiste (partly hidden), Susan Jennings, Don Godwin, Bill Greenstreet, and Walt Sartory.*

*Current and retired members of the Pressure Vessel Technology Section turned out in 1989 to wish long-time secretary, Sue Freels, a happy retirement. Pictured from left are Bill Pennell, Lynn Crawley, Dan Naus, Dan Curtis, Jack Smith, Tom King, Barry Oland, Sam Bolt, Marge Taylor, Claud Pugh, Freels, Fred Jackson, Sandra (Birch) Kennedy, Dick Cheverton, Bob Bryan, Marshall McFee, Grover Robinson, Jeff Parrott, Tim Theiss, Terry Dickson, Hubert Guinn, Charlie Hurtt, and Bob Smith.*

*Division secretaries also held a luncheon for Sue Freels at her retirement. Seated from left are unidentified, Becky Harrell, Freels, and Marge Taylor. Standing from left are Sharon Fuller, Lynn Crawley, Bettye Seivers, Becky Fortner, Tammy Narramore, Debbie (Bailey) Milsap, Teresa Leonard, Fay Duncan, Amy Leslie, Delores Weaver, Julia Cox, Jean Fraley, Saylor Webb, Gwen Scudder, Linda Dockery, and Jean Bray.*

*Ray Hudson probably holds the division record in terms of years of company service at retirement—45 years and 7 months! Pictured here at his 1990 retirement party are (seated from left) Lara James, Brenda Williams, Hudson, Claire Luttrell, and Florence Olden; (standing) John Jones, Dan O'Connor, Bruce Poole, Frank Swinson, Tammra Horning (partially hidden), John Merkle, Rick Battiste, Don Godwin, Susan Jennings, Tina Phillips, unidentified, Perk Cooper, Walt Sartory, Don Williams, Terry Yahr, Joe Blass, Sam Moore, Fay Duncan, Grover Robinson, Jim Corum, Bill Hendrich, Joe Pidkowicz, (retired DOE employee), and Taka Ogata (assignee from CRIEPI in Japan.)*

*The Nuclear Operations Analysis Center staff posed for this picture at Joel Buchanan's retirement in 1991. Kneeling from left are Ernie Silver, Andrea Cross, Ron Thornton, and Don Copinger. Standing, from left, are Joe Minarick, Jim Rooney, Mike Muhlheim, Buchanan, George Murphy, Angie (Puckett) Redford, Jana Hammonds, Linda Kerekes, Chuck Mitchell, Debbie Queener, Mike Poore, John Farquharson, Gwen Scudder, Leonard Palko, Bill Kohn, Joe Cletcher, Gary Mays, Marge Fish, Mike Plaster, and Ralph Guymon.*



*In 1978, President Jimmy Carter became the first president to visit ORNL while in office. In the Building 4500N lobby, Herman Postma introduced him to John Jones, who explained a fluidized-bed coal burner designed to cogenerate power and heat. Secretary of Energy, James Schlesinger, looks on as the president shakes hands with Jones.*

*Several members of the division clerical staff have just received training certificates in this 1979 photograph. Shown with Bob MacPherson are, seated from left, Kathleen Emch, Debbie Hendrix, Mary Phillips, Nancy Markham, and Roberta Poe; standing are Gwen (Talley) Scudder and Carole (Sweeden) Kappelmann.*

*Division members of the Energy Systems Inventor's Forum (patent holders) posed for this 1985 picture. Seated from left are Grady Whitman, Mac Lackey, Howard Bowers, Roy Huddleston, and Dick Cheverton. Standing are Tom Cole, Bob MacPherson, Stuart Daw, Dave Thomas, Colin West, Dave Eissenberg, Reg McCulloch, Bill Greenstreet, Tom Kress, and Ralph Dial.*

*The Heavy-Section Steel Technology Program, NRC's longest running R&D activity, for many years sponsored meetings of a Vessel Integrity Review Group. In 1988, all four individuals who had served as program managers were in attendance. They are (1) Joel Witt, lower right; (2) Grady Whitman, lower left; (3) Claud Pugh, upper right; and (4) Bill Corwin, upper left. Bill Pennell currently serves as the fifth manager.*

# 6. SUMMARY

In its nearly 50 years of existence, the Engineering Technology Division (ETD) has evolved from an organization under the Manhattan District, shrouded in secrecy and focused on nuclear pilot plant work for producing and separating small amounts of plutonium into a sophisticated, nationally and internationally known, engineering organization that successfully conducts a broad spectrum of highly technical activities for a number of clients. The clients, of course, include DOE, which likewise evolved from the Manhattan District organization. This transition was driven by pursuit of nuclear reactor development for many varied applications, starting from isotope production and the investigation of nuclear irradiation effects on materials through the development of power-producing reactors for civilian and military use. Breeder reactors in which fuel is produced simultaneously with power were studied and tested, and large-scale reactors have been examined for use in agro-industrial complexes to support and enhance the quality of life in poverty-stricken areas of the world.

With the advent of nuclear power use for the generation of electricity on a commercial scale, the focus broadened to include safety aspects, environmental effects, and general consequences of widespread nuclear energy use. These considerations required a broadened approach to nuclear power and greatly expanded expertise to deal with the multifarious aspects. This need for broadened perspective was met in a purposeful fashion that allowed intellectual and professional growth to meet the organizational and technical challenges associated with nuclear energy and to address energy production, conservation, and use in general. This approach has increased the stature of the organization and positioned it to contribute to technology development on a wide front, in keeping with the thrust to make ORNL a national resource in the broadest sense.

When formed in 1944, the Technical Division consisted of about 50 members. In 1969, the Reactor Division reached a high of 288; ETD had 186 members in May 1992. The events of the last 50 years were unprecedented and unpredictable. The potential for the next 50 years is equally exciting. A consistent characteristic of the Division over the years is that, through excellence and perseverance, it has overcome adversity, turned problems into opportunities, and established a reputation for thriving on change.

# Appendix A

# KEY PERSONNEL

**Table A.1. Division Directors**

| Dates | Division Director | Associate,[a] Assistant,[b] Technical,[c] and Deputy[d] Director |
|---|---|---|
| | **Technical Division** | |
| 1944–1948 | M. C. Leverett | |
| 1948–1949 | M. D. Peterson | J. A. Lane[a]<br>F. L. Steahly[a]<br>C. E. Winters[a] |
| 1949–1950 | A. M. Weinberg | |
| | **Reactor Technology Division** | |
| 1950–1951 | A. M. Weinberg | M. M. Mann[a] |
| | **Reactor Experimental Engineering Division** | |
| 1951–1953 | C. E. Winters | |
| 1953–1955 | J. A. Lane | |
| 1955–1957 | J. A. Lane | R. B. Briggs[a]<br>E. G. Bohlman[b] |
| 1957–1958 | J. A. Lane | E. G. Bohlman[b]<br>R. N. Lyon[b] |
| 1958–1960 | R. B. Briggs | E. G. Bohlman 1958[b], 1959–60[a]<br>R. N. Lyon 1958[b], 1959–1960[a] |
| | **Aircraft Nuclear Propulsion Division** | |
| 1951 | R. C. Briant | |
| 1952 | R. C. Briant | J. H. Buck[a]<br>A. J. Miller[b] |
| 1953–1954 | R. C. Briant | A. J. Miller |
| | **Aircraft Reactor Engineering Division** | |
| 1954–1958 | S. J. Cromer | |
| | **Reactor Projects Division** | |
| 1958–1959 | W. H. Jordan | A. L. Boch[a]<br>A. P. Fraas[a]<br>H. G. MacPherson[a]<br>A. J. Miller[b] |

**Table A.1 (continued)**

| Dates | Division Director | Associate,[a] Assistant,[b] Technical,[c] and Deputy[d] Director |
|---|---|---|
| 1959 | R. A. Charpie | A. L. Boch[a] |
| | | A. P. Fraas[a] |
| | | H. G. MacPherson[a] |
| | | A. J. Miller[b] |
| 1959–1960 | R. A. Charpie | A. L. Boch[a] |
| | | A. P. Fraas[a] |
| | | A. J. Miller[b] |
| | **Reactor Division** | |
| Late 1960 | R. A. Charpie | |
| 1961 | R. A. Charpie | C. E. Winters[d] |
| 1961–1963 | H. G. MacPherson, Acting | S. E. Beall[b] |
| | | A. P. Fraas[a] |
| | | R. N. Lyon[a] |
| 1963–1971 | S. E. Beall | A. P. Fraas[a] |
| | | R. N. Lyon[a] |
| 1971–1974 | S. E. Beall | A. P. Fraas[a] |
| | | R. N. Lyon[c] |
| 1974–1977 | G. G. Fee | |
| | **Engineering Technology Division** | |
| 1977–1978 | G. G. Fee | |
| 1978–1989 | H. E. Trammell | |
| 1989–1992 | J. E. Jones Jr. | W. R. Martin[a] |

**Table A.2. Section or Department Heads and Project or Program Leaders**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| 1946 | Process Development | M. D. Peterson |
| (Leverett) | Engineering Development | R. B. Briggs |
| | Process Design | J. R. Huffman |
| | Engineering Materials | J. A. Kyger |
| | Engineering Research | R. N. Lyon |
| | Technical Operations | W. A. Rodger |
| 1948 | Process Development | F. L. Steahly |
| (Peterson) | Engineering Development | R. B. Briggs |
| | Process Design | J. R. Huffman |
| | Engineering Materials | J. A. Kyger |
| | Engineering Research | R. N. Lyon |
| | Pilot Plant | D. G. Reid, Acting |
| 1949 | Process and Pile Design | W. R. Gall |
| (Peterson) | Engineering Development | C. B. Graham |
| | Engineering Materials | R. N. Lyon, Acting |
| | Engineering Research | R. N. Lyon |
| | Chemical Process Development | F. L. Steahly |
| | Pilot Plants | D. G. Reid |
| 1949 | Process and Mechanical Design | J. A. Lane |
| (Weinberg) | Reactor Physics | E. P. Blizard |
| | Engineering Research and Development | C. E. Winters |
| | Project | M. M. Mann |
| | Chemical Development | F. L. Steahly |
| 1950 | Analysis and Design | J. A. Lane |
| (Weinberg) | Engineering Research and Development | C. E. Winters |
| | MTR Project | M. M. Mann |
| | ANP Project | A. M. Weinberg |
| | **Reactor Experimental Engineering Division** | |
| 1951 | Projects | M. M. Mann |
| (Weinberg) | Engineering Research and Development | C. E. Winters |
| 1951 | Engineering Research | R. N. Lyon |
| (Winters) | Engineering Development | C. B. Graham |
| | 7500 Area | S. E. Beall |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| | HR Controls | L. R. Quarles |
| | HR Design | W. R. Gall |
| 1952 | Engineering Research | R. N. Lyon |
| (Winters) | Engineering Development | C. B. Graham |
| | 7500 Area | S. E. Beall |
| | HR Corrosion | E. G. Bohlmann |
| | HR Controls | W. M. Breazeale |
| | HR Design | R. B. Briggs |
| 1952 | HR Engineering Research | R. N. Lyon |
| (Winters) | HR Engineering Development | C. B. Graham |
| | Heat Transfer and Hydrodynamics | H. F. Poppendiek |
| | Homogeneous Reactor Experiment | S. E. Beall |
| | HR Corrosion | E. G. Bohlmann |
| | HR Instruments and Controls | W. M. Breazeale |
| | HR Design | R. B. Briggs |
| 1953 | HR Engineering Research | R. N. Lyon |
| (Lane) | HR Engineering Development | C. B. Graham |
| | Heat Transfer and Hydrodynamics | H. F. Poppendiek |
| | Homogeneous Reactor Experiment | S. E. Beall |
| | HR Corrosion | E. G. Bohlmann |
| | HR Instruments and Controls | |
| | HR Design | R. B. Briggs |
| | Reactor Physics | M. C. Edlund |
| 1954 | Engineering Research | R. N. Lyon |
| (Lane) | Engineering Development | C. B. Graham |
| | Heat Transfer and Hydrodynamics | H. F. Poppendiek |
| | Homogeneous Reactor Tests | S. E. Beall |
| | Instruments and Controls | J. N. Baird |
| | Corrosion | E. G. Bohlmann |
| | Design | R. B. Briggs |
| | Reactor Analysis | M. C. Edlund |
| 1955 | Engineering Research | R. N. Lyon |
| (Lane) | Engineering Development | C. B. Graham |
| | Heat Transfer and Hydrodynamics | H. F. Poppendiek |
| | Homogeneous Reactor Tests | S. E. Beall |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| | Instruments and Controls | W. P. Walker |
| | Corrosion | E. G. Bohlmann |
| | Design | R. B. Briggs |
| | Reactor Analysis | M. C. Edlund |
| 1955 | Engineering Research | R. N. Lyon |
| (Lane) | Engineering Development | C. B. Graham |
| | Heat Transfer and Physical Properties | H. F. Poppendiek |
| | Homogeneous Reactor Test (HRT) | S. E. Beall |
| | Instrumentation and Controls | W. P. Walker |
| | Corrosion Studies | E. G. Bohlmann |
| | Mechanical Design | W. R. Gall |
| | Process Design | R. B. Korsmeyer |
| | Reactor Analysis | P. R. Kasten |
| 1956 | Engineering Research | R. N. Lyon |
| (Lane) | Engineering Development | C. B. Graham |
| | Heat Transfer and Physical Properties | H. F. Poppendiek |
| | HRT | S. E. Beall |
| | Instrumentation and Controls | D. S. Toomb |
| | Corrosion Studies | E. G. Bohlman |
| | Mechanical Design | W. R. Gall |
| | Process Design | R. B. Korsmeyer |
| | Reactor Analysis | P. R. Kasten |
| 1957 | Design | W. R. Gall |
| (Lane) | HRT | S. E. Beall |
| | Engineering Development | I. Spiewak |
| | Reactor Analysis | P. R. Kasten |
| | Engineering Research | R. B. Korsmeyer |
| | Instrumentation and Controls | D. S. Toomb |
| | Materials Research | E. G. Bohlmann |
| 1958 | Design and Engineering | R. N. Lyon |
| (Briggs) | HRT | S. E. Beall |
| | Reactor Materials Research | E. G. Bohlmann |
| 1959 | Component Development | I. Spiewak |
| (Briggs) | Design | W. R. Gall |
| | Research and Analysis | P. R. Kasten |
| | Systems Development | R. B. Korsmeyer |

Table A.2 (continued)

| Date | Project or Program Section or Department | Name |
|---|---|---|
| | HRT | S. E. Beall |
| | Solution Materials | J. C. Griess |
| | Slurry Materials | E. L. Compere |
| | Materials Experiment Engineering | H. C. Savage |
| | Solution Materials Radiation | G. H. Jenks |
| | **Aircraft Nuclear Propulsion/Aircraft Reactor Engineering/Reactor Projects Division** | |
| 1951 | Central Design | C. B. Ellis |
| (Briant) | Experimental Engineering | H. W. Savage |
| | ANP Physics | N. M. Smith, Jr. |
| | General Design | C. B. Ellis |
| | | A. P. Fraas |
| | Experimental Engineering | H. W. Savage |
| | ARE Project | W. M. Breazeale |
| 1952 | ANP Physics | W. K. Ergen |
| (Briant) | General Design | A. P. Fraas |
| | Experimental Engineering | H. W. Savage |
| | ARE Project | E. S. Bettis |
| 1953 | ANP Physics | E. K. Ergen |
| (Briant) | General Design | A. P. Fraas |
| | Experimental Engineering | H. W. Savage |
| | ARE | E. S. Bettis |
| | | J. L. Meem |
| 1954 | Physics | W. K. Ergen |
| (Cromer) | Power Plant Engineering | A. P. Fraas |
| | Design | H. C. Gray (P and W) |
| | Experimental Engineering | H. W. Savage |
| | ARE | E. S. Bettis |
| | | J. L. Meem |
| 1955 | Physics | W. K. Ergen |
| (Cromer) | Power Plant Engineering | A. P. Fraas |
| | Design | H. C. Gray (P and W) |
| | Experimental Engineering | H. W. Savage |
| | Reactor Construction | E. S. Bettis |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| 1955 | Physics | A. M. Perry |
| (Cromer) | Power Plant Engineering | A. P. Fraas |
| | Design | H. C. Gray (P and W) |
| | Experimental Engineering | H. W. Savage |
| | Reactor Construction | E. S. Bettis |
| | 7503 Area Construction | W. G. Piper |
| 1956 | Physics | A. M. Perry |
| (Cromer) | Power Plant Engineering | A. P. Fraas |
| | Design | H. C. Gray (P and W) |
| | Experimental Engineering | H. W. Savage |
| | 7503 Construction | W. G. Piper |
| 1957 | Physics | A. M. Perry |
| (Cromer) | Power Plant Engineering | A. P. Fraas |
| | Design | E. J. Breeding |
| | Experimental Engineering | H. W. Savage |
| | 7503 Construction | W. G. Piper |
| | Heat Transfer and Physical Properties | H. W. Hoffman |
| 1957 | Physics | A. M. Perry |
| (Jordan) | Power Plant Engineering | A. P. Fraas |
| | Design | E. J. Breeding |
| | Experimental Engineering | H. W. Savage |
| | 7503 Construction | W. F. Boudreau |
| | | M. Bender |
| | Heat Transfer and Physical Properties | H. W. Hoffman |
| 1958 | Experimental Engineering | H. W. Savage |
| (Jordan) | Engineering Design | E. J. Breeding |
| | Gas-Cooled Reactor Experiment | F. H. Neill |
| | Army Package Power Reactor and Maritime Ship Reactor | H. C. McCurdy |
| | Physics | A. M. Perry |
| | Engineering Research | H. W. Hoffman |
| | Thermodynamics and Fluid Mechanics | W. T. Furgerson |
| | Industrial Liaison | M. Bender |
| | Special Problems | W. B. Cottrell |
| | Heat Exchangers and Plant Design | J. Foster |
| | Applied Mechanics and Stress Analysis | W. L. Greenstreet |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| 1959 | Gas-Cooled Reactor Program | R. A. Charpie |
| (Charpie) | Army Package Power and Maritime Ship Reactor | A. L. Boch |
| 1959 | Experimental Engineering and Small Reactors | A. L. Boch |
| (Charpie) | Advanced Concepts | A. P. Fraas |
| | ANP Coordination | A. J. Miller |
| | Gas-Cooled Reactors | R. A. Charpie |
| | Journal of Nuclear Safety | W. B. Cottrell |
| 1960 | Experimental Engineering | A. L. Boch |
| (Charpie) | Reactor Design | A. P. Fraas |
| | ANP Coordination | A. J. Miller |
| | Gas-Cooled Reactors | R. A. Charpie |
| | Technical Progress Review of Nuclear Safety | W. B. Cottrell |
| | **Reactor Division** | |
| 1961 | Reactor Design | A. P. Fraas |
| (Charpie) | Reactor Analysis | A. M. Perry |
| | Engineering Development—A | I. Spiewak |
| | Engineering Development—B | H. W. Savage |
| | Engineering Science | R. N. Lyon |
| 1961 | Reactor Operation | S. E. Beall |
| | Radiation Engineering | D. B. Trauger |
| | Special Problems | A. P. Fraas |
| | ANP Coordination | A. J. Miller |
| | Gas-Cooled Reactor Program | R. A. Charpie |
| | Fluid Fuels Reactor Program | R. B. Briggs |
| | High Flux Reactor Project | C. E. Winters |
| 1962 | Reactor Design | M. Bender |
| (MacPherson) | Reactor Analysis | A. M. Perry |
| | Engineering Development—A | I. Spiewak |
| | Engineering Development—B | H. W. Savage |
| | Engineering Science | R. N. Lyon |
| | Reactor Operation | S. E. Beall |
| | Radiation Engineering | D. B. Trauger |
| | Special Problems | A. P. Fraas |
| | Gas-Cooled Reactors | W. D. Manly |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| | High Flux Reactor | A. L. Boch |
| | Molten-Salt Reactors | R. B. Briggs |
| | Space Reactors | A. J. Miller |
| | Thorium Utilization Program | J. A. Lane |
| 1963 | Reactor Design | M. Bender |
| (MacPherson) | Reactor Analysis | A. M. Perry |
| | Engineering Development—A | I. Spiewak |
| | Engineering Development—B | H. W. Savage |
| | Engineering Science | R. N. Lyon |
| | Reactor Operation | S. E. Beall |
| | Radiation Engineering | D. B. Trauger |
| | Special Problems | A. P. Fraas |
| 1964 | Reactor Design | M. I. Lundin |
| (Beall) | Reactor Analysis | A. M. Perry |
| | Engineering Development—A | I. Spiewak |
| | Engineering Development—B | H. W. Savage |
| | Engineering Science | R. N. Lyon |
| | MSRE Operations | P. N. Haubenreich |
| | Irradiation Engineering | H. C. McCurdy |
| | Special Projects | A. P. Fraas |
| 1966 | Reactor Design | M. I. Lundin |
| (Beall) | Reactor Analysis | A. M. Perry |
| | Engineering Development | I. Spiewak |
| | Experimental Engineering | R. E. MacPherson |
| | Engineering Science | R. N. Lyon |
| | MSRE Operations | P. N. Haubenreich |
| | Irradiation Engineering | H. C. McCurdy |
| | Special Projects | A. P. Fraas |
| 1971 | Design | M. I. Lundin |
| (Beall) | Analysis | A. M. Perry |
| | Engineering Development | I. Spiewak |
| | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Nuclear Safety Projects | W. B. Cottrell |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| 1973 | Engineering Analysis | I. Spiewak |
| (Beall) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Nuclear Safety Information Center and Nuclear Safety Journal | W. B. Cottrell |
| 1974 | Engineering Analysis | I. Spiewak |
| (Fee) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Nuclear Safety | W. B. Cottrell |
| | Fast Reactor Safety and Core Systems Programs | M. H. Fontana |
| 1976 | Engineering Analysis | I. Spiewak |
| (Fee) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Safety Information | W. B. Cottrell |
| | Fast Reactor Safety and Core Systems Programs | M. H. Fontana |
| | **Engineering Technology Division** | |
| 1978 | Engineering Analysis | I. Spiewak |
| (Trammell) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Safety Information | W. B. Cottrell |
| | Advanced Reactor Systems | M. H. Fontana |
| 1979 | Engineering Analysis | I. Spiewak |
| (Trammell) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Safety Information | W. B. Cottrell |
| | Advanced Concepts Development | M. H. Fontana |
| | Fossil Energy Technology | J. E. Jones Jr. |
| 1979 | Engineering Analysis | I. Spiewak |
| (Trammell) | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |

**Table A.2 (continued)**

| Date | Project or Program Section or Department | Name |
|---|---|---|
| | Solid Mechanics | G. D. Whitman |
| | Safety Information | W. B. Cottrell |
| | Advanced Concepts Development | M. H. Fontana |
| | Fossil Energy Technology | J. E. Jones |
| 1981 (Trammell) | Engineering Analysis | I. Spiewak |
| | Experimental Engineering | R. E. MacPherson |
| | Heat Transfer—Fluid Dynamics | H. W. Hoffman |
| | Solid Mechanics | G. D. Whitman |
| | Safety Studies | W. B. Cottrell |
| | Advanced Concepts Development | M. H. Fontana |
| | Fossil Energy Technology | J. E. Jones |
| 1982 (Trammell) | Engineering Analysis | J. E. Jones |
| | Experimental Engineering | R. E. MacPherson |
| | Thermal Systems Technology | H. W. Hoffman |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | G. D. Whitman |
| | Nuclear Operations Analysis Center | W. B. Cottrell |
| 1984 (Trammell) | Engineering Analysis | J. E. Jones |
| | Experimental Engineering | R. E. MacPherson |
| | Thermal Systems Technology | H. W. Hoffman |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | G. D. Whitman |
| | Nuclear Operations Analysis Center | J. R. Buchanan |
| 1985 (Trammell) | Engineering Analysis | W. G. Craddick |
| | Experimental Engineering | D. W. Burton |
| | Thermal Systems Technology | H. W. Hoffman |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | G. D. Whitman |
| | Nuclear Operations Analysis Center | J. R. Buchanan |
| 1986 (Trammell) | Engineering Analysis | W. G. Craddick |
| | Experimental Engineering | D. W. Burton |
| | Thermal Systems Technology | H. W. Hoffman |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | C. E. Pugh |
| | Nuclear Operations Analysis Center | J. R. Buchanan |

Table A.2 (continued)

| Date | Project or Program Section or Department | Name |
|---|---|---|
| 1987 | Engineering Analysis | E. C. Fox |
| (Trammell) | Applied Systems Technology | D. W. Burton |
| | Thermal Systems Technology | H. W. Hoffman |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | C. E. Pugh |
| | Nuclear Operations Analysis Center | J. R. Buchanan |
| | Advanced Neutron Source Project Office | C. D. West |
| 1989 | Engineering Analysis | E. C. Fox |
| (Jones) | Applied Systems Technology | D. W. Burton |
| | Thermal Systems Technology | W. G. Craddick |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | R. D. Cheverton |
| | Nuclear Operations Analysis Center | J. R. Buchanan |
| | Space and Defense Technology Program | W. R. Martin |
| 1991 | Engineering Analysis | E. C. Fox |
| (Jones) | Applied Systems Technology | T. S. Kress |
| | Thermal Systems Technology | W. G. Craddick |
| | Structural Mechanics | J. M. Corum |
| | Pressure Vessel Technology | R. D. Cheverton |
| | Nuclear Operations Analysis Center | G. T. Mays |
| | Space and Defense Technology Program | W. R. Martin |

# Appendix B

## ORGANIZATION CHARTS

# TECHNICAL DIVISION ORGANIZATION CHART

3-1-48

M. C. Leverett - Director — Ethel Tuttle, Sec.; Louise Bond, Sec.

- W. M. Hawkins, Assistant Director, Administration — Eleanor Pippin, Sec.; Evelyn Gilliland, Clerk; Jane Snow, Clerk
  - T. W. Hungerford, Assistant to Director, Projects & Safety — Eunice Greenway, Sec.
  - J. R. Whitlock, Administration Assistant, Procurement — Virgil Reynolds, Clerk; June Hale, Clerk
- Stuart McLain, Special Assistant — Jean Kuna, Sec.
- M. D. Peterson, Chief Section T-I, Process Development — Marie Wolfe, Sec.; Agnes Hair, Typist; Ann Leinart, Sec.
  - C. W. Schersten, Assistant to Chief
  - C. D. Watson, Assistant to Chief
  - W. K. Eister, Assistant Chief - Semi-Works
  - F. L. Steahly, Assistant Chief - Laboratory and Plant Assistance
  - D. E. Ferguson, R. E. Leuze, Group Leaders, 23 Extraction — J. L. Bamberg, Tech.; G. C. Blalock, Tech.; R. C. Lovelace, Tech.; H. F. Soard, Tech.
    - (J. W. Gost) (C. V. Ellison)
  - A. C. Jealous, F. R. Bruce, Group Leaders, 25 Extraction — R. O. Chambers, Tech.; G. B. Dinsmore, Tech.; J. E. Farmer, Tech.; G. R. Guinn, Tech.; Guy Johns, Tech.; W. H. Luster, Tech.; F. L. Rogers, Tech.; W. E. Shockley, Tech.; J. W. Smith, Tech.
    - F. N. Browder
    - J. O. Davis
    - E. O. Nurmi
    - G. A. West
    - W. A. Horne - Shift Supervisor
    - C. D. Hylton - Shift Supervisor
    - G. S. Parker - Shift Supervisor

F. R. Bruce, Group Leader, 25 Extraction

- R. E. Blanco
- Arlene Kibbey
- W. B. Lanham
- L. P. Morse
- A. T. Gresky - Special Assignment

A. Johnson, Janitress
C. A. Clark, Tech.
L. A. Byrd, Tech.
W. B. Howerton, Tech.
Vannesse Orr, Tech.
E. R. Jones, Tech.
D. Q. White, Tech.

M. R. Poston, F. N. Browder, Group Leaders
Solvents

B. I. Bailey, Tech.
Gladys Howser, Tech.

F. L. Steahly

- D. C. Overholt
- T. C. Runion

J. M. DeLozier, Tech.
V. L. Fowler, Tech.
R. B. Quincy, Tech.

W. K. Eister

- I. R. Higgins
- J. B. Ruch
- C. D. Watson
- G. A. West

T. D. Napier, Tech.

R. B. Briggs, Chief, Section T-II
Engineering Development

Thelma Sutton, Sec.

S. B. Beall, Leader - Control Elements Group

- T. H. Mauney

A. L. Davis, Tech.
J. J. Hairston, Tech.

J. Reed, Leader - Corrosion Group

W. Kirkland, Tech.
J. L. Stepp, Tech.

O. Sisman, Leader - Pile Irradiation
Engineering Assistant Group

R. L. Townes, Tech.

R. Van Winkle, Leader - Scale Formation and
Water Treatment Group

- C. D. Bopp
- J. B. Chrisney

C. M. Burchell, Tech.
W. B. Krick, Tech.
R. Smith, Tech.

W. B. Allred, Leader - Strength of Materials Group

H. C. Savage

W. H. Stromquist - Special Problem — G. H. Johnstone, Tech.
C. E. Clifford - Special Problem
C. P. Coughlen - Special Problem

(B. W. Kinyon) - Research Shops Coordinator

J. R. Huffman, Chief, Section T-III - Process Design — Mary Dougher, Sec.

C. F. West, Jr. - Administrative Assistant

D. Nicoll, Assistant Chief

C. E. Winters, Associate Section Chief

A. D. Mackintosh

J. A. Lane, Associate Section Chief

R. M. Jones, Joint Leader, Group A
Pile Proper

N.E. Hill
D. Nicoll
S. Scott, Jr.

C. W. Day, Draftsmans
R. C. Allerbe, Draftsman
A. S. Ludlow, Draftsman
H. W. Watts, Draftsman
(C. A. Roberts), Draftsman
(Sue Eatherly), Clerk

J. T. Weills, Joint Leader, Group A, Pile Proper

W. S. Farmer
W. G. Stockdale

G. Hovorka, Leader, Group B - Pile Buildings

F. C. McCullough, Leader, Group C - External Systems

W. R. Gall, Leader, Group D - Pile Mockup

D. J. Mallon
J. R. McWherter
R. A. Long
W. E. Unger
F. C. Zapp

L.
F. M. Culler, Leader, Group E - 1200 - 1300 Areas

- G. Hanson
- H. E. Goeller
- R. L. Klotzbach
- R. P. Milford

J. A. Kyger, Chief, Section IV
Engineering Materials

Ruby Bullard, Sec.
Susan Cornish, Sec.

W. L. Cockrell, Assistant Chief

- F. A. Kocur - Individual Assignment
- C. D. Smith - Leader, Group A
  - G. M. Adamson
  - H. Wallace

  F. Blackshere, Tech.
  C. F. Cutcher, Tech.
  J. H. Day, Tech.
- F. W. Drosten, Leader, Group B
  - J. T. Howe
  - D. A. Lawson
  - V. L. McKinney

  J. N. Hix, Tech.
- F. Kerze, Leader, Group C
  - G. M. Carlton
  - J. E. Cunningham
  - W. H. Wilson

  C. C. Cooley, Tech.
- J. L. English, Leader Group D
  - A. R. Olsen
  - S. H. Wheeler

  R. N. Tench, Tech.
- T. Rockwell, Leader, Group E
  - F. J. Roehrenbeck

(J. H. Erwin)

(Harry Seaman), Machinist

R. N. Lyon, Chief, Section V - Engineering Research

C. C. Hurtt, Tech.

D. G. Reid, Chief Supervisor, Acting Chief — Jeanne Doran, Sec.
Section VI - Pilot Plant

| | |
|---|---|
| H. M. Feder, Assistant Chief Supervisor | R. F. Benson, Operator |
| | H. S. Caldwell, Operator |
| H. K. Jackson | H. G. Chambers, Operator |
| | G. Davis, Operator |
| M. Levenson, Assistant Chief Supervisor | L. L. Fairchild, Operator |
| | C. A. Gifford, Operator |
| K. K. Kennedy, Senior Supervisor | F. M. Grizzell, Operator |
| | J. H. Groover, Operator |
| W. H. Carr | W. Jenings, Operator |
| | C. H. Jones, Operator |
| E. L. Nicholson, Senior Supervisor | J. Land, Operator |
| | W. E. Ledbetter, Operator |
| G. Sadowski | J. F. Lockmiller, Operator |
| | T. R. McLellon, Operator |
| N. J. Rigstad, Senior Supervisor | J. R. Ogle, Operator |
| | R. P. Purkey, Operator |
| B. B. Harrington | M. Richardson, Operator |
| | H. E. Sexton, Operator |
| A. M. Rom, Senior Supervisor | E. E. Shields, Operator |
| | B. J. Stradder, Operator |
| E. M. Shank | G. Tipton, Operator |
| | H. C. Thompson, Operator |
| | J. T. Wiggins, Operator |

(W. P. Bigler - Argonne)

---

Hired: Monthly - Nurmi
Weekly - Bond

Terminated: Monthly - Bigler, Bornwasser, Burris, DeHaan, Ward
Weekly - Allen (January) Caraglin

Transferred In: Leinard from Purchasing Department (Weekly)

Personnel on loan to the Technical Division from other departments are shown in parenthesis.

| | | 2-4-48 | 3-1-48 |
|---|---|---|---|
| Monthly | (Technical) | 105 | 101 |
| Weekly | (Non-technical) | 87 | 88 |
| | | 192 | 189 |

# OAK RIDGE NATIONAL LABORATORY
## REACTOR EXPERIMENTAL ENGINEERING DIVISION

SEPTEMBER 1, 1954

**J. A. LANE - DIRECTOR**
D S SMITH[D]-Secretary

**F. A. KOCUR - Assistant to Director**
D S SMITH[D]-Secretary
M A MEADOWS - Librarian
V H HATCH[*]-Secretary

**W B PIKE - Building Engineer**
W D BUCHANAN
M J PHILLIPS[D] Stenographer

### HEAT TRANSFER AND PHYSICAL PROPERTIES
H.F. POPPENDIEK - Section Chief
T.S CARLSMITH[D]-Secretary

J.O BRADFUTE - Hydrodynamics
L.D. PALMER - Hydrodynamics
D.C. HAMILTON - Free Convection
F.E. LYNCH - Free Convection
H.W. HOFFMAN - Salt Heat Tran.
J. LONES - Salt Heat Tran.
M.W. ROSENTHAL - Tran. Boil.
R.L. MILLER - Tran. Boil.
S.I. COHEN - Den. and Visc.
T.N. JONES - Den. and Visc.
N.D. GREENE - Elec. Cond.
W.D. POWERS - Ht. Cap. & Therm. Cond.
G.C. BLALOCK - Ht Cap.
R.M. BURNETT - Therm. Cond.
S.J. CLAIBORNE - Therm. Cond.
G.M. WINN - Gen. Ht. Trans.

### ENGINEERING DEVELOPMENT
C.B. GRAHAM - Section Chief
G O BROWN - Secretary

**FUEL CIRCULATING PUMPS HEAT EXCHANGER-BLOWERS**
W.L. ROSS
L F GOODE
R.E. WASCHER
R.J. KEDL
K.E. ESTES[MC]
J G SMITH
C.C. HURTT
A.B. DAFFRON

**HRT MOCKUP-RECOMBINERS GAS SEPARATORS-CORES**
I. SPIEWAK
J.A. HAFFORD
P.H. HARLEY[MC]
J.W. SKILLERN
D.M. EISSENBERG
L.W. ROANE[MC]
R. GOODMAN
A.L. JOHNSON

**4000 GPM LOOP-FEED PUMPS H.R. DUMP TEST-MISC.**
J.S. CULVER
C.H GABBARD
B.A HANNAFORD[MC]
E.L. YOUNGBLOOD[MC]
R G. MEZA
J.M. BAKER
S.R WEST
W.B KRICK

### CORROSION STUDIES
E. G. BOHLMANN[CD] Section Chief
M.E McCLUER - Secretary

**DATA FILE**
R E WACKER

**LABORATORY SERVICE CORROSION TESTS**
E.L. COMPERE[CD]
J L ENGLISH
G E MOORE[CD]
S.H WHEELER
H L BARKER
R. F. BENSON
L L FAIRCHILD
J L UNDERWOOD
B J WARD

**DYNAMIC LOOP TESTS**
J.C. GRIESS
R S GREELEY
S.A REED
W H HINDS
P R SHIRE
R M WALLER

H.C. SAVAGE
W.N BLEY
N C BRADLEY[MC]
F J WALTER
R.M. WARNER
J F WINESETTE - Ldr.
L.L ALLEN
C.R BENSON
F H BROOME
C.M BURCHELL
J.B FRENCH
H F HENDERSON
J E JACKSON
F E McDANIEL
F N MORGAN
L G PHELPS
R H PLEMONS
R. G. SHERROD
A.L. SUTTON
W.H. WAGNER

**IN-PILE LOOPS**
G H JENKS[CD]
J E BAKER
D T JONES[EM]
V R BULLINGTON
R A LORENZ
J W BROWN
E H GUINN
J W MYERS
B W. SCOTT
A R OLSEN
B W McCOLLUM

### HRT
S.E. BEALL - Section Chief
M R SUMNER - Secretary

S. VISNER
S.I. KAPLAN
G.H JOHNSTONE
J.J. HAIRSTON[A]
R SMITH, Jr
J D JONES
J.W. HILL
P.M. WOOD

### HR METALLURGY
E.C MILLER[MD] Section Chief
F L LOVING - Secretary

G E ELDER[MD]
J I FEDERER[MD]
W J FRETAGUE[MD]
W O HARMS[MD]
W J LEONARD[MD]
G B WADSWORTH[MD]
C H WODTKE[MD]
T W FULTON, Jr.[MD]
J J WOODHOUSE[MD]

### REACTOR DESIGN
R.B. BRIGGS - Section Chief
A.L. GUNNELS[D] Secretary

W R GALL - Asst. Section Chief
M B HOY - Secretary
R E BROOKS - Clerk

**HRT DESIGN**

PROCESS ANALYSIS
R. VanWINKLE
R E AVEN

MECHANICAL DESIGN
R.H. CHAPMAN
R.D. CHEVERTON
T H MAUNEY
R.G PITKIN[C]
R.C. ROBERTSON[C]
C L SEGASER
F.C. ZAPP

PLANT LAYOUT AND DRAFTING
T.H. THOMAS
W. ROBINSON
W. TERRY
R L CAUBLE
C W. DAY
T E HAYNES
A S LUDLOW
C.A ROBERTS
R.D. SATTERFIELD
R.H. SIGLER
W T STERLING
J.A WATTS

STRUCTURE & SERVICES
J. N. ROBINSON

**TBR DESIGN****

PROCESS ANALYSIS
P N HAUBENREICH
R. McGRATH[C]

MECHANICAL DESIGN
A L GAINES
M. I. LUNDIN
J R McWHERTER
W F TAYLOR[C]
H H BRADFORD

**ORR DESIGN*****
J P GILL - Project Engr
R. HENGSTLER - Secretary

PROCESS ANALYSIS
F T BINFORD
N.F LANSING
J.P SANDERS

MECHANICAL DESIGN
A W TELL[SSD]
H L WATTS
W.L WRIGHT
J.E ROBERTS

**SLIDES & REPORT DWGS**
M E ARNOLD
J.W MILLSAPS

### INSTRUMENTATION & CONTROL
J.N. BAIRD - Section Chief
M.J. PHILLIPS[D]-Stenographer

**HRT INSTRUMENTATION**
W P WALKER
A.M. BILLINGS[ICD]
R.L. MOORE[ICD]
R.E. TOUCEY[ICD]
K W WEST[ICD]

**DESIGN & CONTROL ROOM LAYOUT**
E. VINCENS[RC]
M.C. BECKER[RC]
J. R. BROWN[RC]
P. T. HURAY[RC]

**HRP SUPPORTING FACILITIES**
D.S. TOOMB
D.G. DAVIS[ICD]
J.L. REDFORD

### REACTOR ANALYSIS & LONG RANGE PLANNING
M.C. EDLUND - Section Chief
A L GUNNELS[D] Secretary

H.C. CLAIBORNE
J.P. DILORENZO
L. DRESNER
T.B. FOWLER
P.R. KASTEN
L.C. NODERER
M TOBIAS

### ENGINEERING RESEARCH
R.N. LYON - Section Chief
T.S. CARLSMITH[D] Secretary

P.C. ZMOLA - Boiling Studies
M. RICHARDSON - Exptl. Studies
H. A. McCOLL
R. V. BAILEY[C]

A.S. KITZES - Slurry Properties and Components
R. B. GALLAHER - Heat Transfer
P. R. CROWLEY - Mockup Coord.
B. A. KRESS - Pressurizer
W. O. HULLING - Pumps
C. A. GIFFORD
C. G. LAWSON - Press. Vessel Studies

L.F. PARSLY - Slurry Blanket Mockup
R.H. NIMMO
L. E. McTAGGART
W. B. PLUMMER

D.G. THOMAS - Colloidal Properties of Slurries
J. O. PERRET
P. H HAYES
S. R. BUXTON[AC]

**PERSONNEL SUMMARY**

| DIVISION EMPLOYEES (9-1-54) | TECHNICAL | TECHNICIANS | STAFF | LOANEES | TOTAL |
|---|---|---|---|---|---|
| ADMINISTRATION | 0 | 0 | 6 | 0 | 6 |
| HEAT TRANSFER & PHYSICAL PROPERTIES | 10 | 7 | 0 | 0 | 17 |
| HRT | 6 | 3 | 1 | 0 | 10 |
| CORROSION STUDIES | 12 | 29 | 1 | 7 | 49 |
| ENGINEERING DEVELOPMENT | 12 | 8 | 1 | 5 | 26 |
| REACTOR DESIGN | 23 | 11 | 6 | 4 | 44 |
| INSTRUMENTATION & CONTROLS | 3 | 0 | 1 | 10 | 14 |
| ENGINEERING RESEARCH | 15 | 3 | 1 | 1 | 20 |
| METALLURGY | 0 | 0 | 1 | 10 | 11 |
| REACTOR ANALYSIS & LONG RANGE PLANNING | 8 | 0 | 0 | 0 | 8 |
| TOTAL | 89 | 61 | 18 | 37 | 205 |

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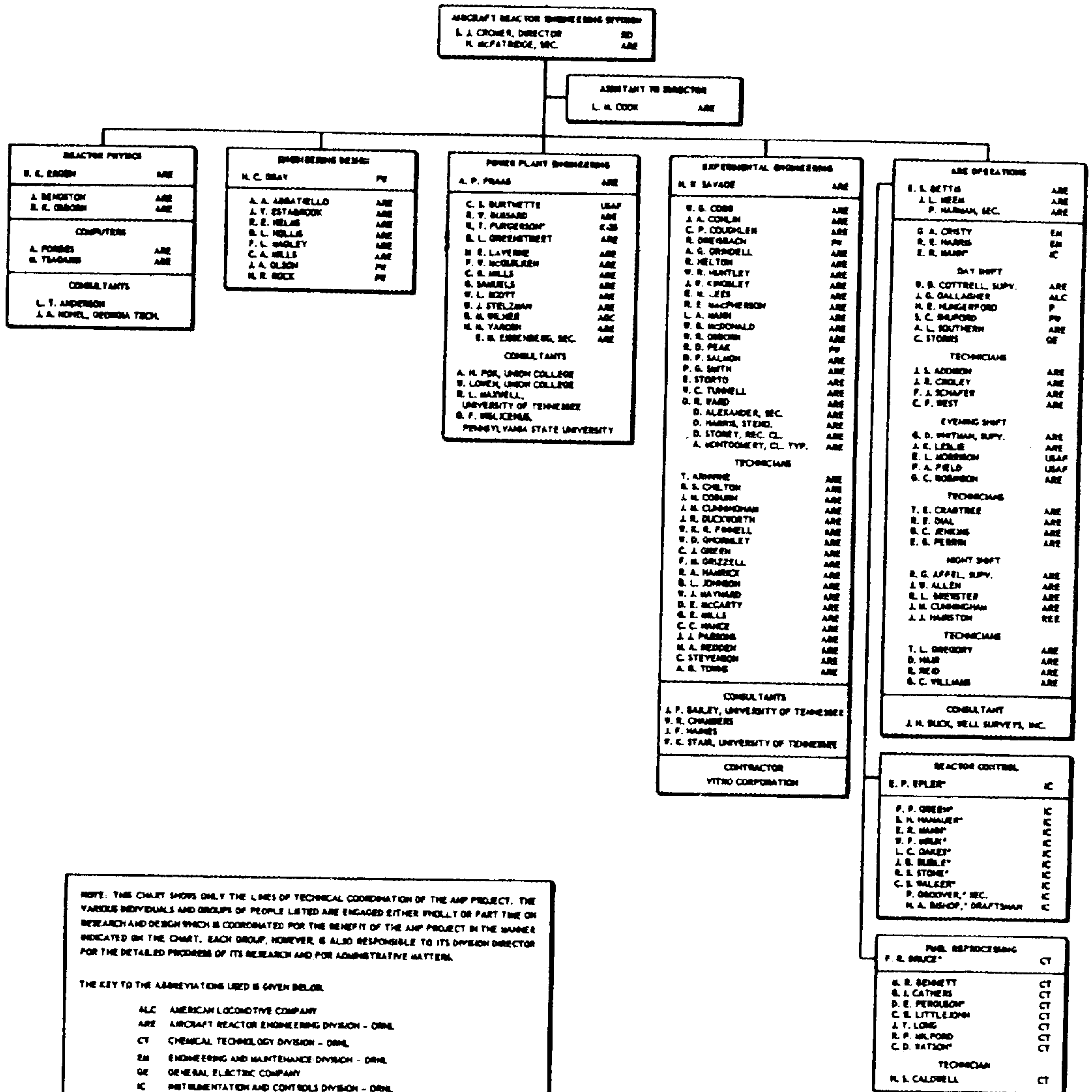
**REFERENCE**

RC ON LOAN FROM REACTOR CONTROLS
A ON LOAN TO ARE
AC ON LOAN FROM ANALYTICAL CHEMISTRY DIVISION
C CONSULTANT
CD ON LOAN FROM CHEMISTRY DIVISION
D DUAL CAPACITY
EM ON LOAN FROM ENGINEERING & MAINTENANCE DIVISION
ICD ON LOAN FROM INSTRUMENTATION & CONTROLS DIVISION
MC ON LOAN FROM MATERIALS CHEMISTRY DIVISION
MD ON LOAN FROM METALLURGY DIVISION
RDD RESEARCH DIRECTOR'S DIVISION
SSD ON LOAN FROM SOLID STATE DIVISION

*ON LOAN TO W.C. MOORE
**SUPERVISION BY W.C. MOORE, RDD
***SUPERVISION BY T. E. COLE, ICD




NOTE: THIS CHART SHOWS ONLY THE LINES OF TECHNICAL COORDINATION OF THE ANP PROJECT. THE VARIOUS INDIVIDUALS AND GROUPS OF PEOPLE LISTED ARE ENGAGED EITHER WHOLLY OR PART TIME ON RESEARCH AND DESIGN WHICH IS COORDINATED FOR THE BENEFIT OF THE ANP PROJECT IN THE MANNER INDICATED ON THE CHART. EACH GROUP, HOWEVER, IS ALSO RESPONSIBLE TO ITS DIVISION DIRECTOR FOR THE DETAILED PROGRESS OF ITS RESEARCH AND FOR ADMINISTRATIVE MATTERS.

THE KEY TO THE ABBREVIATIONS USED IS GIVEN BELOW.

| | |
|---|---|
| ALC | AMERICAN LOCOMOTIVE COMPANY |
| ARE | AIRCRAFT REACTOR ENGINEERING DIVISION - ORNL |
| CT | CHEMICAL TECHNOLOGY DIVISION - ORNL |
| EM | ENGINEERING AND MAINTENANCE DIVISION - ORNL |
| GE | GENERAL ELECTRIC COMPANY |
| IC | INSTRUMENTATION AND CONTROLS DIVISION - ORNL |
| P | PHYSICS DIVISION - ORNL |
| PW | PRATT AND WHITNEY AIRCRAFT DIVISION - UAC |
| RD | RESEARCH DIRECTOR'S DEPARTMENT - ORNL |
| USAF | UNITED STATES AIR FORCE |

*PART TIME

SEPTEMBER 1, 1954

OAK RIDGE NATIONAL LABORATORY
REACTOR DIVISION

AUGUST 1, 1963

DIVISION ORGANIZATION CHART

S. E. BEALL,[1] DIRECTOR
A. P. FRAAS,[1] ASSOC. DIRECTOR
R. N. LYON,[1] ASSOC. DIRECTOR

- H. M. BECKLER,[3] BUDGETS
- A. W. SAVOLAINEN, REPORTS — PAGE 8
- ANALYSIS — A. M. PERRY — PAGE 2
- DESIGN — M. BENDER — PAGE 3
- ENGINEERING DEVELOPMENT - A — I. SPIEWAK — PAGE 4
- ENGINEERING DEVELOPMENT - B — H. W. SAVAGE — PAGE 5
- ENGINEERING SCIENCE — R. N. LYON — PAGE 6
- IRRADIATION ENGINEERING — D. B. TRAUGER — PAGE 7
- OPERATIONS — S. E. BEALL,[1] ACTING — PAGE 8
- SPECIAL PROJECTS — A. P. FRAAS[1] — PAGE 9
- ENGINEERING AND ADMINISTRATIVE SERVICES — W. R. OSBORN — PAGE 10

STAFF ASSIGNMENTS

HFIR PROJECT
T. E. COLE

TECHNICAL PROGRESS REVIEW OF NUCLEAR SAFETY
W. B. COTTRELL,[1] EDITOR

SPECIAL RESEARCH PROBLEMS
W. K. ERGEN

SPACE PROGRAM COORDINATION
A. J. MILLER[2]
JEAN WHITE

[1]DUAL CAPACITY
[2]RADIATION CONTROL OFFICER
[3]ON ASSIGNMENT FROM BUDGET AND PROGRAMMING

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**REACTOR ANALYSIS**
A. M. Perry
Margaret Wilson

**HIGH-FLUX REACTOR**
H. C. Claiborne[1]

**MOLTEN-SALT REACTOR**
P. N. Haubenreich[4]
J. R. Engel
B. E. Prince

**EXPERIMENTAL GAS-COOLED REACTOR**
C. A. Preskitt
E. A. Nephew
J. V. Wilson

**SPACE REACTOR PROGRAM**
P. H. Pitkanen
O. L. Smith

**REACTOR THEORY AND NUMERICAL METHODS**
T. W. Kerlin
M. L. Tobias

**CODE DEVELOPMENT**
T. B. Fowler
J. L. Lucius - ORGDP
M. E. Tsagaris
C. D. Griffies
H. A. MacColl

**LONG-RANGE PLANNING**
M. W. Rosenthal[1]
Shirley Kirkwood

- **THORIUM REACTOR EVALUATION**
  L. G. Alexander
  W. L. Carter[3]
  C. W. Craven
  R. Van Winkle

- **REACTOR EVALUATION STUDIES**
  M. W. Rosenthal[1]
  L. L. Bennett
  E. S. Bettis[2]
  H. C. Claiborne[1]
  E. H. Gift
  D. R. Vondy

- **ADVANCED GAS-COOLED REACTOR**
  R. S. Carlsmith
  H. F. Bauman
  J. G. Delene
  W. E. Thomas

**FOREIGN ASSIGNMENT**
P. R. Kasten
Julich, Germany

**SPECIAL ASSIGNMENT**
R. D. Cheverton HFIR

[1]Dual Capacity
[2]On Loan From Design Department
[3]On Loan From Chem. Tech.
[4]On Loan From Operations Department

**REACTOR DESIGN**
M. BENDER
W. R. GALL,[1] ASSISTANT
MARION HOY

**HFIR DESIGN**
J. R. MCWHERTER
LOUISE FERGUSON
N. HILVETY
H. A. MCLAIN
R. E. SCHAPPEL
J. H. WESTSIK

**EGCR PROJECT**
G. D. WHITMAN
VIOLA ERICKSON
M. L. MYERS
G. C. ROBINSON
R. D. STULTING

**ADVANCED GCR DESIGN**
W. R. GALL[1]
T. G. CHAPMAN
R. E. HOSKINS

**MPRE DESIGN**
R. C. OLSON

**CONSULTANT**
R. C. DANIELS

**SPECIAL ASSIGNMENT**
R. S. VALACHOVIC HFIR
R. C. ROBERTSON MSRE

**ENGINEERING DEVELOPMENT DESIGN**
M. I. LUNDIN
JANICE HUMPHREY

**DESIGN ENGINEERS**
E. J. BREEDING
C. J. CLAFFEY
W. G. COBB
C. W. COLLINS
J. P. GILL
T. L. HUDSON
J. KERR
L. R. KOFFMAN
W. TERRY
L. V. WILSON
F. C. ZAPP
R. FREY[2]

**DESIGN DRAFTING**
CHIEF DRAFTSMAN - A. P. MARQUARDT
ASS'T. CHIEF DRAFTSMAN - C. A. MILLS

| | |
|---|---|
| H. H. BRADFORD | W. B. KRICK |
| W. M. BROWN | F. F. MORRIS |
| J. C. CABLE | J. O. NICHOLSON |
| R. L. CAUBLE | J. J. PLATZ |
| E. E. CHAMBERS | H. M. POLY |
| J. K. DUNCAN | C. A. ROBERTS |
| W. C. GEORGE | W. ROBINSON |
| F. GUNNESS | C. F. SALES |
| T. E. HAYNES | A. E. SELLS |
| G. R. HICKS | R. H. SIGLER |
| S. F. HOWELL | W. G. STERLING |
| C. C. HURTT | J. A. WATTS |
| R. H. JONES | R. L. WEEKS |
| A. R. KERR | |

[1] DUAL CAPACITY
[2] ON LOAN FROM Y-12 ENGINEERING

**ENGINEERING DEVELOPMENT - A**
I. Spiewak
Gerry Brown

**PROGRAM COORDINATORS**

| | |
|---|---|
| R. M. Hill[1,4] | HFIR |
| C. G. Lawson[1] | EVAL. |
| J. C. Moyers[1] | ATR |
| D. Scott[1] | MSRE |

**HFIR DEVELOPMENT**
R. M. Hill[1,4]
G. J. Dixon[5]
J. D. Hutchins[5]
J. E. Jones
W. H. Kelley
R. F. Benson
J. P. Hurst
G. M. Winn

**MSRE SYSTEM DEVELOPMENT**
J. L. Crowley
R. B. Gallaher
R. E. Carnes
W. H. Duckworth

**REACTOR MAINTENANCE**
E. C. Hise
R. Blumberg
S. E. Bolt
P. P. Holz[1]
C. Jones
J. R. Shugart

**MSRE COMPONENTS**
D. Scott[1]
M. Richardson
A. N. Smith
H. S. Weber
C. A. Gifford

**FLOW PROBLEMS**
R. J. Kedl
B. J. Young

**THORIUM UTILIZATION PROGRAM**
R. H. Chapman[2]

**NUCLEAR SAFETY PILOT PLANT**
P. P. Holz[1]
L. J. Shersky
C. M. Smith,[6]
G. A. Windle

**REACTOR EVALUATION**
C. G. Lawson[1]
D. M. Eissenberg
J. C. Moyers[1]
H. G. Moore[3]

**CONSULTANTS**
F. N. Peebles
P. N. Stevens

[1]Dual Capacity
[2]On Loan to KEMA, Arnhem, The Netherlands
[3]On Loan To Metals and Ceramics Division
[4]On Loan From Plant and Equipment Division
[5]On Assignment From Operations Division
[6]On Loan From Metals and Ceramics Division

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## ENGINEERING DEVELOPMENT - B

H. W. SAVAGE[1]
R. E. MACPHERSON,[1] ASS'T.
SALLIE JANSCH

### GCR ENGINEERING DEVELOPMENT

F. H. NEILL
- W. F. FERGUSON
- B. C. GARRETT
- D. L. GRAY
- R. E. HELMS
- T. S. KRESS
- M. N. OZISIK[2]
- A. M. SMITH
- E. R. TAYLOR
  - G. E. MILLS
  - J. E. WOLFE

### MSR PUMP DEVELOPMENT

A. G. GRINDELL[1]
- JEWELL HAYES
- C. H. GABBARD[1]
- P. G. SMITH
  - J. M. COBURN

### HIGH-TEMPERATURE MATERIALS PROGRAM

R. E. MACPHERSON[1]
- C. W. CUNNINGHAM
- L. C. FULLER
  - C. G. HENLEY

### SPECIAL ASSIGNMENT

D. W. CARDWELL

### SNAP CORROSION PROGRAM

R. E. MACPHERSON[1]
- W. R. HUNTLEY

### SODIUM TECHNOLOGY

H. W. SAVAGE[1]
- D. B. LLOYD

### SPACE SYSTEMS DEVELOPMENT

J. ZASLER
- JAMIE MASON
- P. A. GNADT

A. G. GRINDELL[1]
- L. W. EVERS
- C. H. GABBARD[1]
- H. C. YOUNG
  - J. M. CUNNINGHAM
  - H. E. PENLAND

### CONSULTANTS

- F. A. ANDERSON
- J. F. BAILEY
- R. B. KNIGHT
- M. W. MILLIGAN
- F. L. SCHWARTZ
- W. K. STAIR
- A. W. CULP

[1] DUAL CAPACITY
[2] ON LOAN FROM SPECIAL PROJECTS DEPARTMENT

**ENGINEERING SCIENCE**
R. N. Lyon[1]
Betty Williams[3]

- **HEAT TRANSFER AND HYDRODYNAMICS**
  H. W. Hoffman, Section Chief
  J. J. Keyes, Assoc. Section Chief
  Dolores Eden
  - **FUNDAMENTAL STUDIES**
    W. R. Gambill
    G. J. Kidd, Jr.
    R. P. Wichner[1]
    R. E. Dial
    J. Lones
    R. M. Summers
  - **SPACE PROGRAM**
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    W. K. Sartory
    J. L. Wantland
    R. L. Miller
    B. J. Sutton
  - **PHYSICAL PROPERTIES**
    J. W. Cooke
    S. J. Claiborne
    U. C. Fulmer, Jr.
  - **SALINE WATER PROGRAM**
    D. G. Thomas
    R. P. Wichner[1]
    P. H. Hayes
  - **GCR PROGRAM**
    R. D. Bundy
    D. H. Wallace
  - **CONSULTANT**
    T. S. Chang
- **APPLIED MECHANICS**
  B. L. Greenstreet, Section Chief
  Delores Weaver
  - **EGCR AND AGR**
    S. E. Moore
  - **GRAPHITE PROGRAM**
    J. G. Merkle[2]
    F. J. Witt
    G. T. Yahr
    D. T. Godwin
  - **SPECIAL PROBLEMS**
    J. M. Corum
  - **STRESS LABORATORY**
    J. E. Smith
    H. D. Curtis
  - **CONSULTANTS**
    R. L. Maxwell
    L. R. Shobe
- **SLIDE PREPARATION AND ACTING LIBRARIAN**
  F. M. Burkhalter

[1]Dual Capacity
[2]On Loan From Plant and Equipment Division
[3]Part Time Loan From Directors Division

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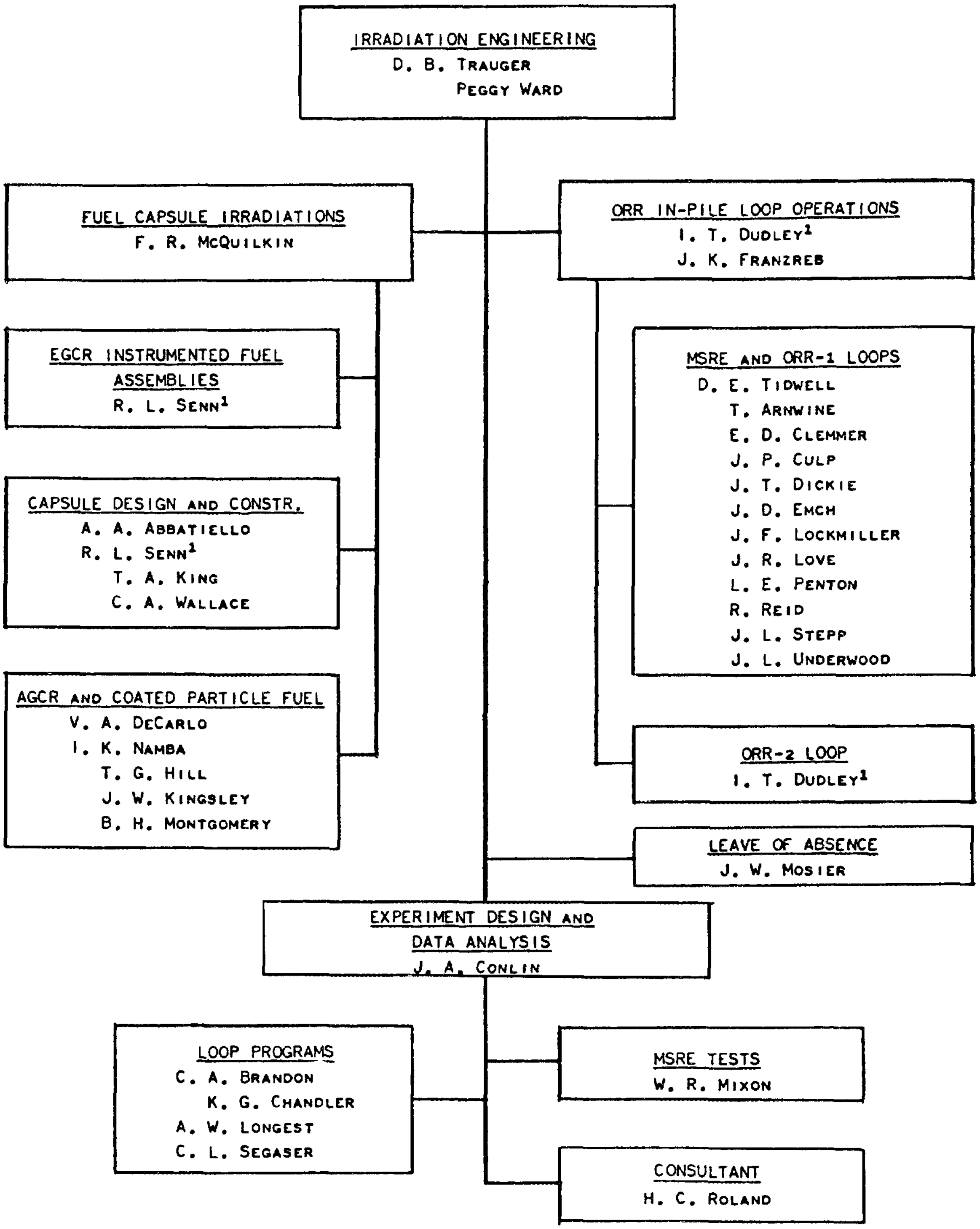
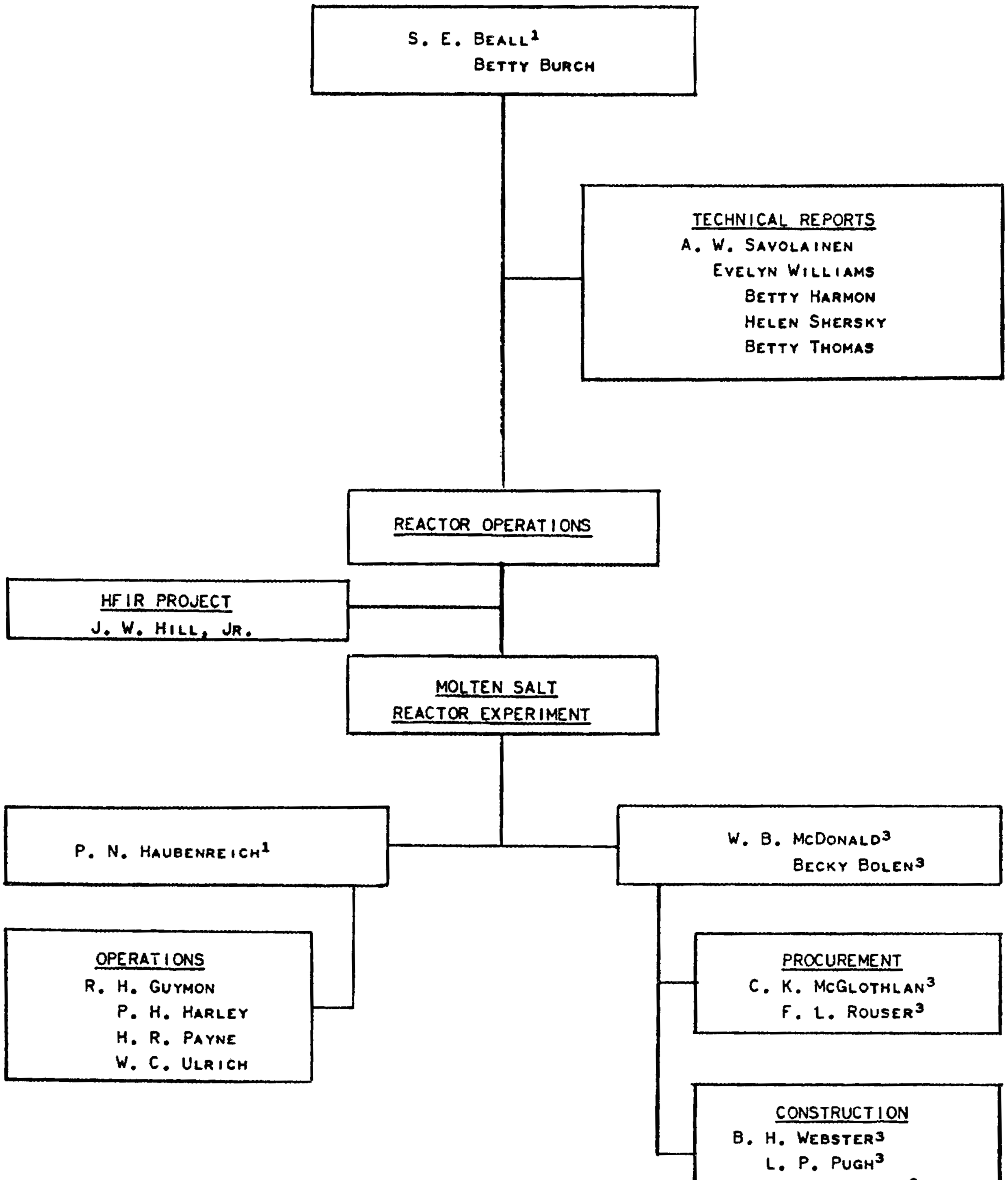



[1]Dual Capacity




[1]Dual Capacity
[2]On Loan to HFIR Project
[3]On Loan to MSRE Project

**SPECIAL PROJECTS**
A. P. Fraas[1]
Shirley Hendrix

**SMALL REACTORS**
H. C. McCurdy
Ellis Stonecipher
T. D. Anderson
V. O. Haynes
S. I. Kaplan
O. H. Klepper
H. C. Roller[3]
L. D. Schaffer
M. L. Winton

**REACTOR SAFETY**
W. B. Cottrell[1]
Margaret Walls
J. R. Buchanan[1]
J. O. Kolb
L. F. Parsly[2]
H. B. Piper
T. H. Row

**NUCLEAR SAFETY INFORMATION CENTER**
W. B. Cottrell[1]
J. R. Buchanan[1]
Juanita Perrou

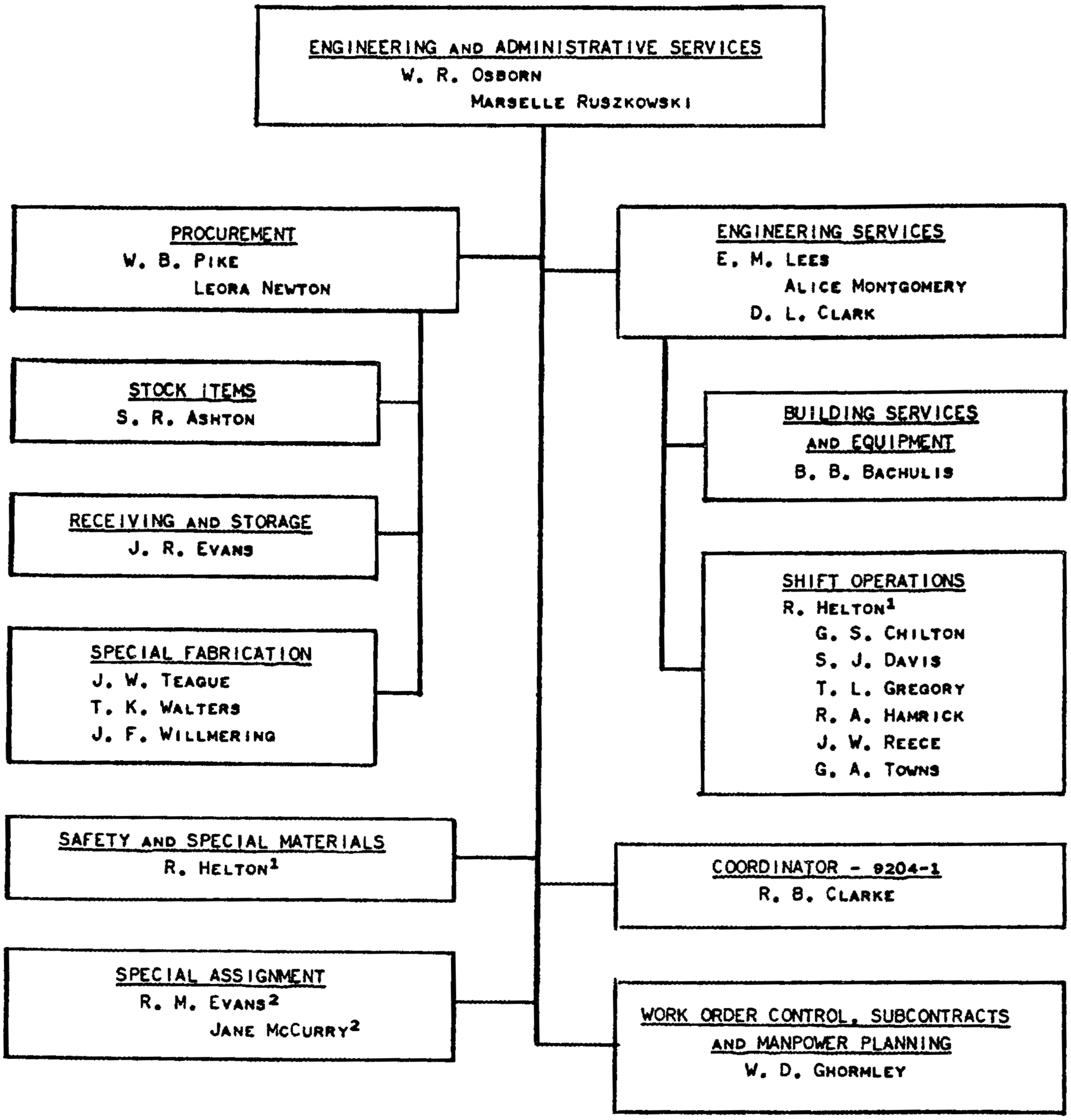
**SPECIAL PROBLEMS**
R. B. Korsmeyer
M. E. LaVerne
M. M. Yarosh

**THERMODYNAMICS AND FLUID DYNAMICS**
G. Samuels
R. S. Holcomb
J. K. T. Jung
M. E. Lackey
J. W. Michel
J. J. Tudor
B. H. Fitzgerald
F. E. Lynch

**POWER PLANT EQUIPMENT**
J. Foster
J. T. Meador
J. K. Jones

**CONSULTANTS**
W. R. Chambers
P. F. Pasqua

[1]Dual Capacity
[2]On Loan From Design Department
[3]On Loan From Engineering Development - B




[1]DUAL CAPACITY
[2]ASSIGNED TO GCR PROJECT DIRECTOR

OAK RIDGE NATIONAL LABORATORY
REACTOR DIVISION

September 1, 1973

S. E. Beall,[1] Director
Betty Burch[1]
R. N. Lyon,[1] Technical Director
Dolores Eden
A. P. Fraas,[1] Assoc. Director

- STAFF ASSIGNMENTS — Page 2
- PROGRAM ASSIGNMENTS — Page 3
- ENGINEERING ANALYSIS — I. Spiewak, A. M. Perry — Page 4
- EXPERIMENTAL ENGINEERING — R. E. MacPherson — Page 5
- HEAT TRANSFER - FLUID DYNAMICS — H. W. Hoffman — Page 6
- SOLID MECHANICS — G. D. Whitman[1] — Page 7
- NUCLEAR SAFETY INFORMATION CENTER & NUCLEAR SAFETY JOURNAL — W. B. Cottrell[1] — Page 8
- ENGINEERING and ADMINISTRATIVE SERVICES — Page 9

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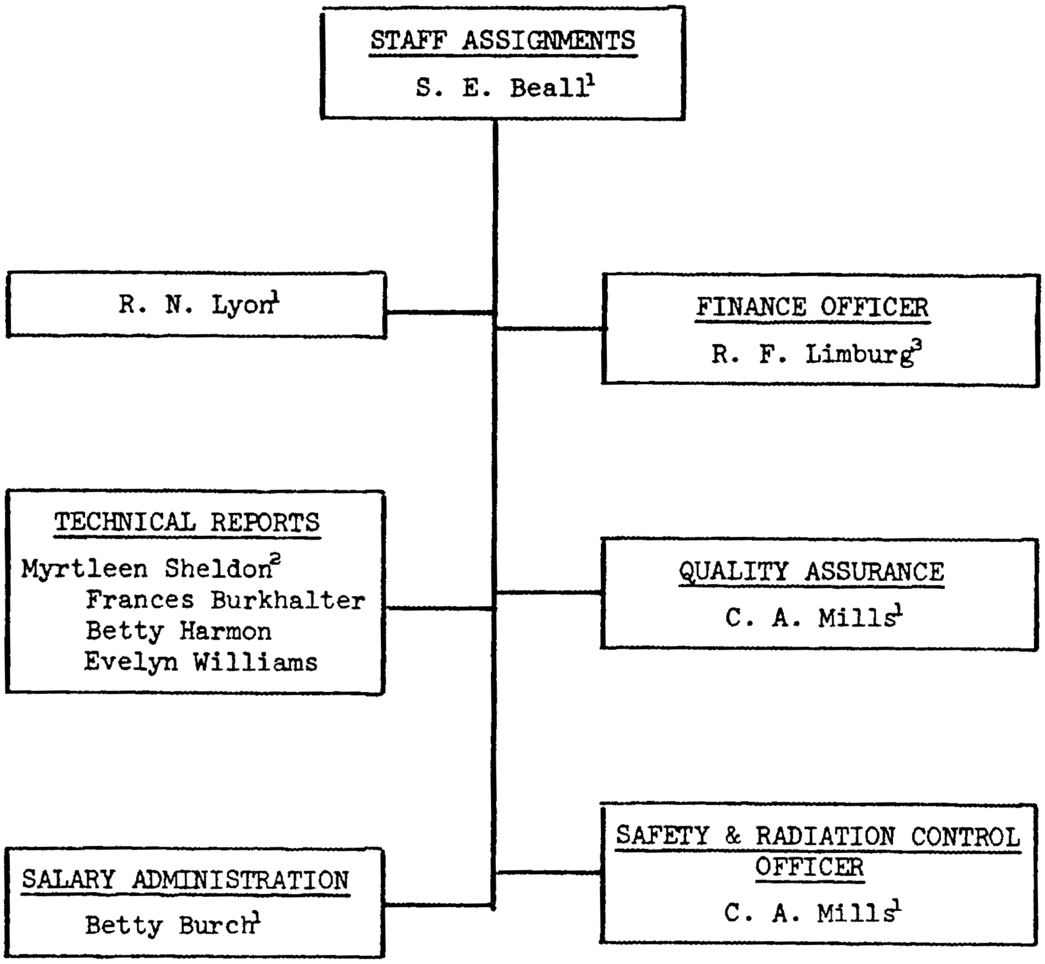


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[2] On loan from Technical Information Div.
[3] On loan from Budget & Programming

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S. E. Beall,[1] Coordinator

**GCR PROGRAMS**

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June Zachary

**LMFBR PROGRAM**

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**PRESSURE VESSEL PROGRAMS**

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Sue Freels
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C. C. Hurtt

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**STUDIES & EVALUATION PROGRAM**

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**ENVIRONMENTAL REPORTS PROJECT**

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T. D. Anderson, Mgr. AEC Program

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A. P. Fraas[1]
Jean Scott
G. Samuels
R. S. Holcomb
M. E. Lackey
M. E. LaVerne[3]
Bernice Fitzgerald
J. J. Tudor

**NUCLEAR SAFETY INFORMATION CENTER & JOURNAL**

W. B. Cottrell,[1] Director

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**SPECIAL ASSIGNMENTS**

C. C. Burwell - Long Range Planning Group
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Annabel Legg - Thermonuclear Div.
S. I. Kaplan[2] - Asst. to D. B. Trauger
C. K. McGlothlan - Quality Assurance Office

[1] Dual Capacity
[2] On loan to Director's Div.
[3] Loan to AEC Regulatory

## ENGINEERING ANALYSIS DEPARTMENT

I. Spiewak[1]
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A. M. Perry,[1] Assoc.
Margaret Wilson

### NUCLEAR DESALINATION PROGRAM

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J. G. Delene, Desalting Controls
J. E. Jones, Desalting Coupling

### HUD-MIUS GROUP

E. C. Hise, Real Site Studies
J. O. Kolb, Environmental Impact
J. T. Meador, Energy Systems
W. R. Mixon, Technology Assessment
C. L. Segaser, Technology Assessment
J. V. Wilson,[1] Computer Models

### REACTOR ANALYSIS SECTION

A. M. Perry[1]
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H. T. Kerr, GCR Cross Sections
B. E. Prince, Reactor Evaluations
W. E. Thomas, Thorium Fuel Recycle
M. L. Tobias, HTGR Safety, CTR, NSIC
J. C. Turnage, Reactor Evaluations

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E. S. Bettis, Consultant

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Louise Ferguson
R. F. Bennett, EDHT Analysis
R. J. Kedl, GCR Safety and Fish Damage Studies
R. Smith

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J. D. Jenkins, Loan to AEC Regulatory
L. Jung, Loan to AEC Regulatory
J. C. Moyers, NSF Program
E. A. Nephew, NSF Program
J. R. Shugart, Loan to Experimental Equipment Engineering
W. E. Thompson, Loan to AEC Regulatory
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G. A. Windle, Loan to Y-12

### CAPITAL COST STUDIES GROUP

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Bettye Seivers
J. A. Conlin,[1] Asst.
A. G. Grindell,[1] Asst.
Virginia Maggart

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J. A. Conlin[1]

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K. R. Thoms[1]
B. H. Montgomery[1]

GCFBR
A. W. Longest

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B. H. Montgomery[1]
L. P. Pugh
C. A. Wallace[1]

**EXPERIMENTAL ASSEMBLY**

C. A. Wallace[1]

**STAFF**

D. L. Clark[1]

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F. E. Lynch - Special Projects
J. E. Wolfe - Heat Transfer-Fluid Dynamics Dept.

**SYSTEMS & COMPONENTS DEVELOPMENT**

A. G. Grindell[1]

**FUEL FAILURE MOCKUP**

P. A. Gnadt
G. E. Mills
H. E. Penland
T. S. Kress
L. F. Parsly
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**ELECTRIC HEATER DEVELOPMENT**

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**THERMAL TRANSIENT TEST**

H. C. Young
W. H. Duckworth

**POTASSIUM TOPPING CYCLE**

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[1] Dual Capacity

## HEAT TRANSFER - FLUID DYNAMICS

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Viola Erickson
D. G. Thomas,[1] Assoc.

### REACTOR THERMAL SAFETY

H. W. Hoffman[1]

**Blowdown Heat Transfer**
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J. L. Crowley
C. G. Lawson
S. Malang[2]
C. D. Griffies

### LMFBR ACTIVITIES

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**FFM Thermal Hydraulics**
D. G. Thomas[1]
P. H. Hayes

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D. G. Thomas[1]

**Reverse Osmosis**
J. D. Sheppard[1]

### REACTOR EFFLUENT THERMAL HYDRAULICS

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H. F. Bauman
H. A. McLain
R. C. Robertson
M. Siman-Tov[3]
F. Vaslow[4]
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R. J. Ryan[7]
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### SPECIAL ASSIGNMENTS

O. H. Klepper, Off-Shore Siting
S. L. Milora, Cold Vapor Technology
W. K. Sartory, MAN Fluid Dynamics, LMFBR Thermal Stress Analysis
J. D. Sheppard,[1] Hydrology

[1] Dual Capacity
[2] Assigned from Karlsruhe Institute for Nuclear Research
[3] Full time from General Engineering Div.
[4] Consultant
[5] Leave of Absence
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[8] On loan from Engineering Analysis Dept.

**SOLID MECHANICS DEPARTMENT**

G. D. Whitman[1]
W. L. Greenstreet,[1] Assoc.
Sue Freels

**Coordinator**

S. E. Moore[1]

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J. E. Smith[1]

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- S. E. Bolt
  - J. P. Rudd

LMFBR Program
- J. E. Smith[1]
  - T. G. Hill
  - K. C. Liu
    - E. H. Guinn
  - M. Richardson
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  - H. D. Curtis
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J. M. Corum,[1] Assoc.
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- R. C. Gwaltney
  - J. W. Bryson
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- S. E. Moore[1]
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  - G. T. Yahr
    - D. T. Godwin[1]
    - H. A. MacColl

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- J. G. Merkle

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- J. P. Callahan[1]

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[1] Dual Capacity
[2] Part Time from Heat Transfer-Fluid Dynamics Dept.

**NUCLEAR SAFETY INFORMATION CENTER**
**and**
**NUCLEAR SAFETY JOURNAL**

W. B. Cottrell[1]
Gwendolyn Woodard[1]

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W. B. Cottrell,[1] Director
J. R. Buchanan,[1] Asst. Director
Lillian Bean
Bobbie Bohanan
Jamie Copeland
Betty Griffith
Debbie Sharp

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| R. B. Gallaher | R. L. Scott[11] |
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| C. F. Holoway[5] | M. L. Winton[7] |
| B. L. Houser[5] | |

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| W. R. Casto | OD |
| J. R. Engel | RD |
| J. C. Hart | HPD |
| E. W. Hagen[6] | I&C |

[1] Dual Capacity
[2] On assignment to Environmental Reports Project
[3] Part time from M&C Division
[4] Part time from Ecological Sciences Division
[5] Part time from Health Physics Division
[6] Part time from I&C Division
[7] Contract
[8] Director's Division
[9] Part time from Heat Transfer-Fluid Dynamics Department
[10] Part time from Pressure Vessel Program
[11] Part time from Operations Division
[12] Part time from the Atmospheric Turbulence & Diffusion Laboratory
[13] Part time from Engineering Analysis Department
[14] Sr. Research Advisor
[15] Part time

**ENGINEERING and ADMINISTRATIVE SERVICES**
S. E. Beall[1]

**ADMINISTRATIVE SERVICES**
Marselle Ruszkowski

**ENGINEERING SERVICES SAFETY and SS MATERIALS, OFFICE MACHINERY**
E. M. Lees
Helen Shersky[1]

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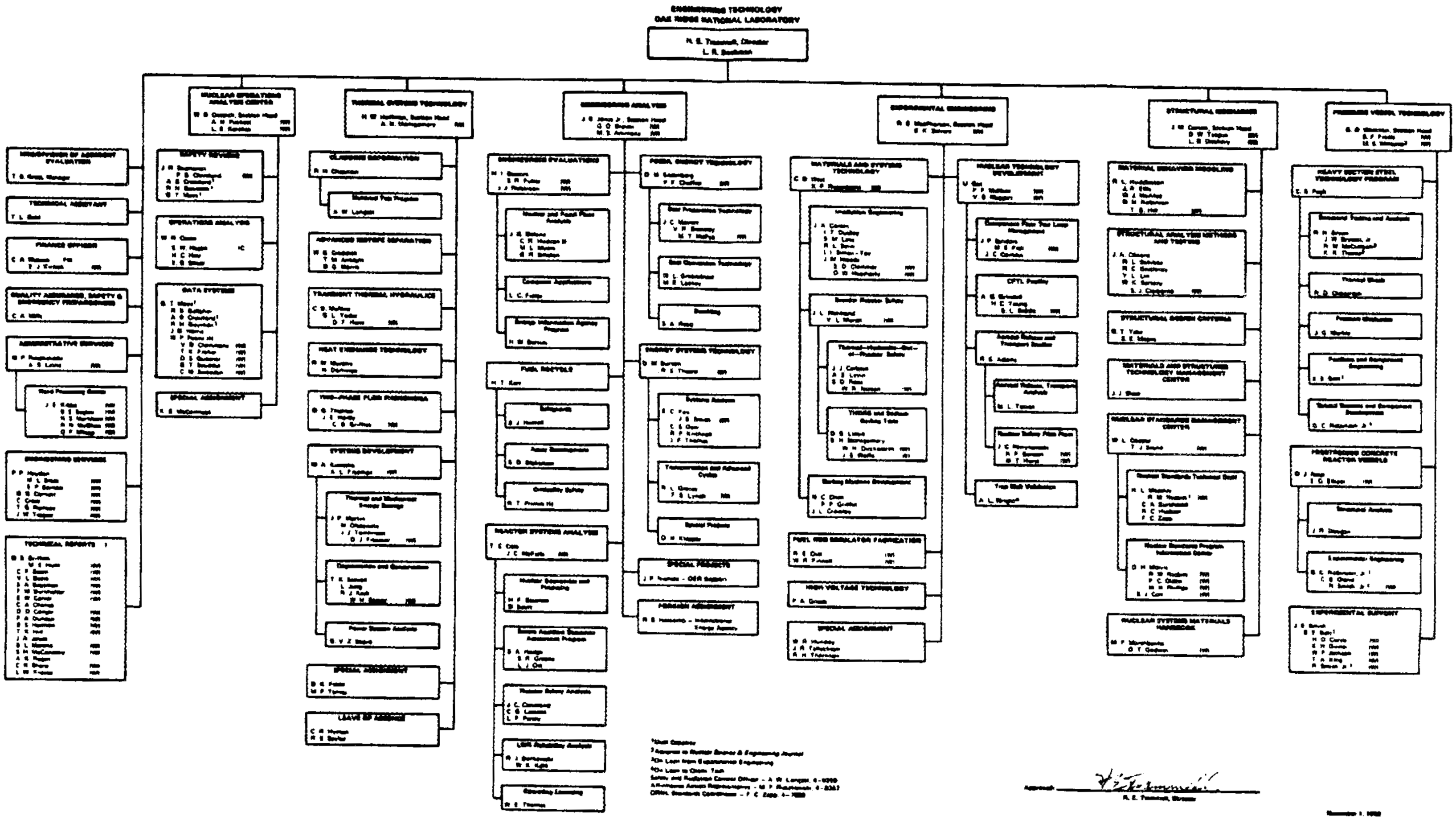
S. R. Ashton[1]
J. W. Teague[1]
T. K. Walters
Helen Shersky[1]

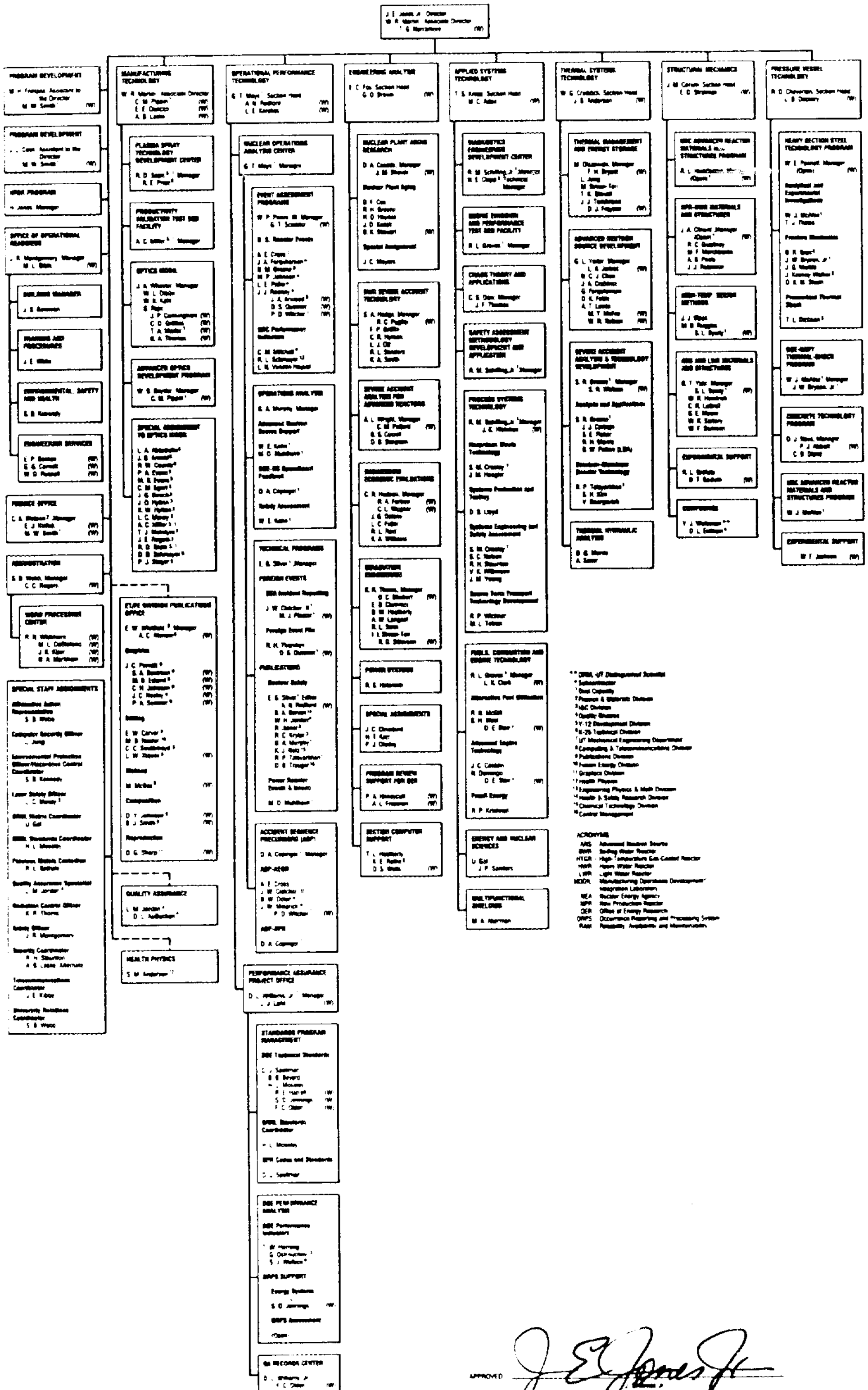
**COORDINATOR - 9204-1**
J. W. Teague[1]

**WORK ORDER CONTROL, MANPOWER PLANNING, RECEIVING & STORAGE**
S. R. Ashton[1]

[1] Dual Capacity


ENGINEERING TECHNOLOGY
OAK RIDGE NATIONAL LABORATORY

**ORNL**
**ENGINEERING TECHNOLOGY DIVISION**
AUGUST 1992




APPROVED J E Jones Jr

## Appendix C

## PHOTOGRAPHS OF 1992 ETD STAFF

*DIVISION OFFICE AND SUPPORT STAFF*

***From left to right, front row: D. L. (Don) AuBuchon, J. R. (Richard) Montgomery, C. A. (Charlie) Watson, G. G. (Jerry) Cornett, E. J. (Judy) Kiriluk, M. L. (Louise) Bible, S. B. (Sandra) Kennedy, M. L. (Michelle) Bryant, J. K. (Jama) Kizer; back row: J. S. (Scott) Bowman, J. E. (John) Jones Jr., L. M. (Larry) Jordan, W. D. (Bill) Russell, M. W. (Michelle) Smith, R. N. (Robin) Whitmore, J. L. (Jack) Cook, C. C. (Chris) Rogers, J. E. (Judy) Kibbe, and M. H. (Mario) Fontana. Not present: S. B. (Saylor) Webb, N. A. (Nancy) Markham, and E. P. (Ed) Benton.***

*OPERATIONAL PERFORMANCE TECHNOLOGY*

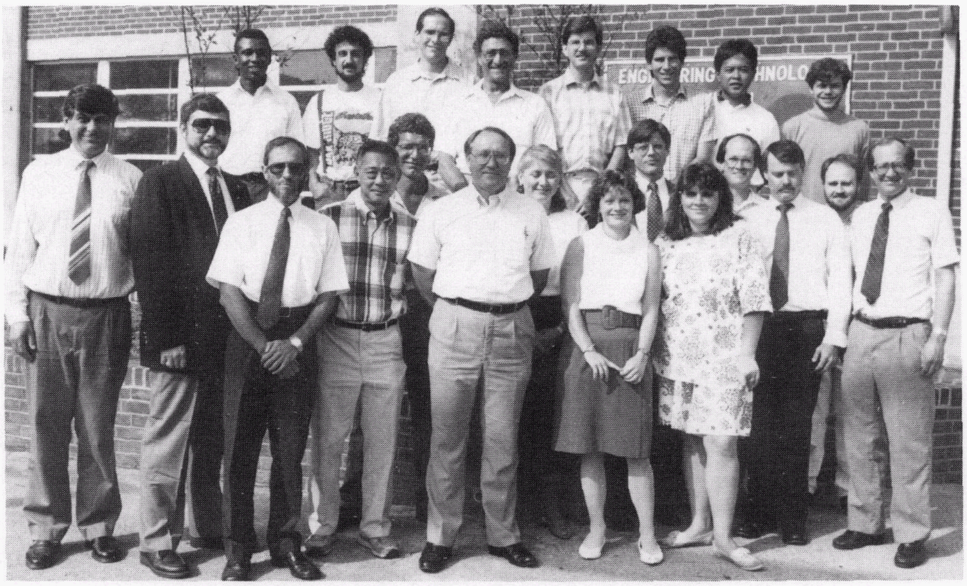
***From left to right, front row: A. R. (Amy) Bush, P. D. (Pam) Witcher, L. J. (Lori) Lane, A. N. (Angie) Redford, D. S. (Debbie) Queener, R. A. (Becky) Harrell, F. C. (Florence) Olden, E. G. (Ernest) Silver, D. J. (Don) Spellman, A. E. (Andrea) Cross, H. L. (Harry) Moseley; back row: G. A. (George) Murphy, R. H. (Ron) Thornton, D. A. (Don) Copinger, J. W. (Joe) Cletcher, B. B. (Bruce) Bevard, W. E. (Bill) Kohn, G. T. (Gwen) Scudder, W. P. (Mike) Poore, M. J. (Mike) Plaster, M. D. (Mike) Muhlheim, and G. T. (Gary) Mays. Not present: T. W. (Tammra) Horning, S. D. (Susan) Jennings, L. E. (Linda) Kerekes, and D. L. (Don) Williams.***

*ENGINEERING ANALYSIS*

***From left to right: P. J. (Pedro) Otaduy, J. C. (John) Moyers, R. G. (Bob) Sitterson, R. S. (Bob) Holcomb, D. W. (Dennis) Heatherly, J. G. (Jerry) Delene, K. A. (Kent) Williams, B. S. (Brian) Cowell, C. L. (Cathy) Wagner, C. M. (Carol) Pollard, R. L. (Ron) Senn, R. A. (Becky) Fortner, B. L. (Becky) Powell, B. K. (Brian) Stewart, R. H. (Becky) Greene, D. F. (Daryl) Cox, R. L. (Bob) Sanders, C. R. (Cliff) Hyman, T. L. (Terry) Heatherly, D. B. (Dave) Simpson, D. A. (Don) Casada, E. C. (Ted) Fox, H. D. (Howard) Haynes, J. D. (John) Kueck, J. M. (Jeanie) Shover, M. D. (Mike) Todd, L. J. (Larry) Ott, R. L. (Lowell) Reid, D. S. (Danny) Walls, N. L. (Nathan) Wood, A. W. (Al) Longest, R. C. (Roxanne) Puglisi, D. C. (Doris) Shubert, A. L. (Tony) Wright, I. I. (Ilana) Siman-Tov, M. A. (Mary) Barto, and F. P. (Fred) Griffin. Not present: G. O. (Jerry) Brown, E. D. (Earl) Clemmer, J. C. (John) Cleveland, A. L. (Angie) Freeman, L. C. (Len) Fuller, S. A. (Steve) Hodge, P. A. (Pat) Honeycutt, C. R. (Randy) Hudson, H. T. (Howard) Kerr, and K. R. (Ken) Thoms.***

*APPLIED SYSTEMS TECHNOLOGY*

***From left to right, front row: N. (Norberto) Domingo, D. M. (David) Frey, V. K. (Van) Wilkinson, M. C. (Margie) Adair, T. S. (Tom) Kress, J. K. (Judith) Hickman, L. K. (Leesa) Clark, S. M. (Sladjana) Crosley, C. S. (Stuart) Daw, J. B. (Johney) Green, M. L. (Mel) Tobias; back row: M. A. (Al) Akerman, J. F. (John) Thomas, R. L. (Ron) Graves, D. B. (Dave) Lloyd, R. P. (Bob) Wichner, J. M. (Joan) Young, B. H. (Brian) West, S. C. (Sam) Nelson, R. H. (Bob) Staunton, R. M. (Robert) Wagner, and J. C. (Jim) Conklin. Not present: R. M. (Bob) Schilling, D. E. (Doug) Blair, U. (Uri) Gat, J. P. (John) Sanders, J. M. (Janet) Hoegler, R. P. (Kris) Krishnan, and R. N. (Ralph) McGill.***



*THERMAL SYSTEMS TECHNOLOGY*

***From left to right, front row:* A. T. (Trevor) Lucas, S. R. (Sherrell) Greene, J. J. (Juan) Carbajo, L. (Lincoln) Jung, M. T. (Marshall) McFee, M. (Mitch) Olszewski, T. K. (Therese) Stovall, L. A. (Lara) James, W. G. (Bill) Craddick, T. H. (Tracy) Bryant, D. K. (Dave) Felde, R. H. (Bob) Morris, J. A. (Allen) Crabtree, S. E. (Steve) Fisher; *back row:* C. (Cornelius) Ejimofor, A. (Ahmet) Sozer, A. E. (Art) Ruggles, M. (Moshe) Siman-Tov, W. R. (Bill) Nelson, D. G. (Dave) Morris, M. (Masanori) Kaminaga, and C. D. (Chris) Davis. *Not present:* J. B. (Jan) Anderson, C. (Charles) Bentley, N. C. (Norbert) Chen, J. (Jack) Dixon, G. L. (Grady) Yoder, Y. (Yousri) Elkassabgi, D. J. (Delmar) Fraysier, V. (Vlad) Georgevich, S. K. (Seok-ho) Kim, K. (Kenneth) Ndoma-Ogar, S. R. (Sonya) Wallace, J. J. (John) Tomlinson, R. P. (Rusi) Taleyarkhan, and K. Y. (Kerri) West.**

*STRUCTURAL MECHANICS*

***From left to right, front row:* S. L. (Sherry) Byerly, E. D. (Darlene) Stratman, J. J. (Julie) Robinson, L. D. (Gretta) Kitchin, Y. J. (Jack) Weitsman, C. L. (Claire) Luttrell, M. B. (Marina) Ruggles, J. J. (Joe) Blass; *second row:* W. K. (Walt) Sartory, D. G. (Dallas) Smith, J. M. (Jim) Corum, A. D. (Andre) Smith, W. F. (Frank) Swinson, R. C. (Richard) Gwaltney, D. T. (Don) Godwin; *back row:* J. H. (Jon) Thompson, D. L. (Don) Erdman, W. R. (Bill) Hendrich, A. B. (Bruce) Poole, G. T. (Terry) Yahr, M. F. (Marty) Marchbanks, and S. E. (Sam) Moore. *Not present:* J. A. (John) Clinard, R. L. (Rick) Battiste, and R. L. (Roy) Huddleston.**


*PRESSURE VESSEL*

*From left to right, front row: R. D. (Dick) Cheverton, T. L. (Terry) Dickson, C. B. (Barry) Oland, D. K. M. (Dave) Shum, B. R. (Richard) Bass, D. J. (Dan) Naus, W. J. (Wally) McAfee; back row: J. W. (John) Bryson, P. J. (Pam) Abbott, A. K. (Anthea) McKaig, L. B. (Linda) Dockery, J. (Janis) Keeney-Walker, and W. F. (Fred) Jackson, Sr. Not present: J. G. (John) Merkle, W. E. (Bill) Pennell, and T. J. (Tim) Theiss.*

*OPTICS TECHNOLOGY*

*From left to right, front row: R. D. (Roland) Seals, A. B. (Amy) Leslie, J. A. (John) Wheeler, K. A. (Kathy) Thomas, J. L. (Jennifer) Mustaleski, E. E. (Emily) Duncan, M. E. (Marty) Elnicki, J. O. (Jim) Hylton, C. C. (Cory) Echols; back row: W. L. (Bill) Drake, S. (Slo) Rajic, J. P. (Joe) Cunningham, W. K. (Keith) Kahl, J. G. (Jack) Gooch, G. T. (Greg) King, and T. A. (Troy) Marlar. Not present: W. R. (Bill) Martin, C. M. (Tina) Pippin, C. M. (Chuck) Egert, P. A. (Paul) Evans, C. D. (Charlie) Griffies, K. W. (Kathy) Hylton, L. C. (Curt) Maxey, A. C. (Art) Miller, J. E. (Jo Ellen) Rogers, W. B. (Bill) Snyder, and P. J. (Phil) Steger.*